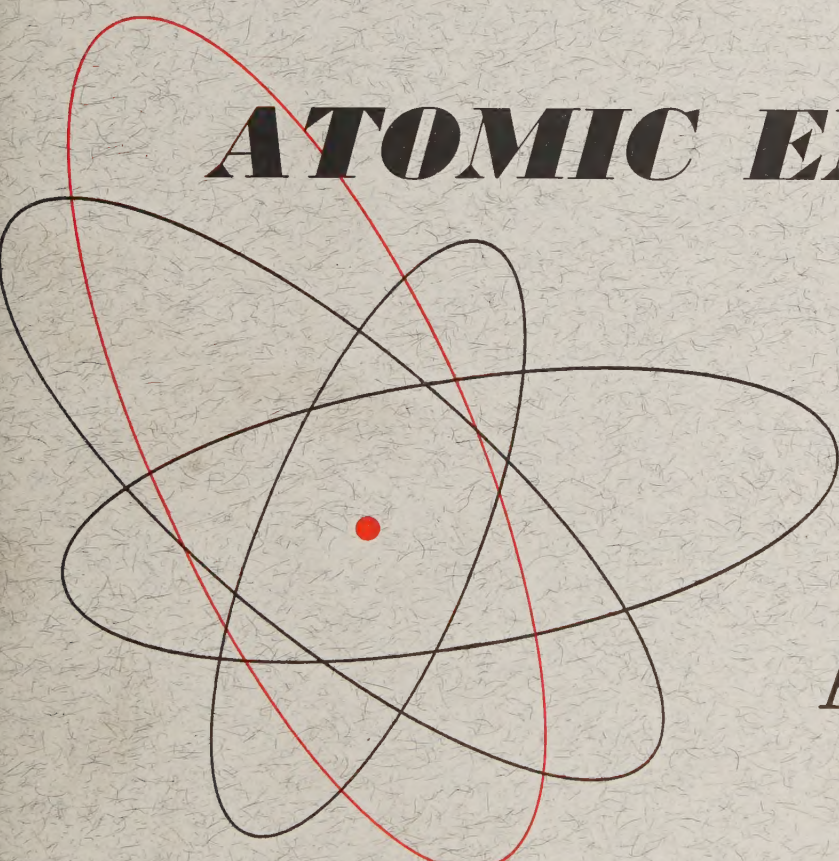


Volume 8, No. 6

July, 1961

THE SOVIET JOURNAL OF

ATOMIC ENERGY



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энергия

TRANSLATED FROM RUSSIAN

CONSULTANTS BUREAU

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- Use of Electron Paramagnetic Resonance for Investigating the Molecular Structure of Coals. N. N. Tikhomirova, I. V. Nikolaeva and V. V. Voevodskii
- New Magnetic Properties of Macromolecular Compounds with Conjugated Double Bonds. L. A. Blyumenfel'd, A. A. Slinkin and A. E. Kalmanson

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Vol. 8, No. 6

July, 1961

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NOTE

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ERRATA

Vol. 8, No. 4, June, 1961

Page	Column	Line	Reads	Should read
277	left	7-8	$\omega_0^2/k^2c^2 \lesssim 1$	$\omega_0^2/k^2c^2 \gtrsim 1$
278	left	14-18	The increase . . .	The increase in the magnitude of the longitudinal velocity component $v_{ } \approx v_0$ to values $v_{ } \approx v_{\perp} > v_0$ means that it is necessary to take account of the additional reduction of $\tau_c(E)$ due to the reduction in the time spent by the ion ($\tau_{ }$) in the region of the heating section: $\tau_{ } \approx 1 \text{ kv}_{ }$.

THE 50-MEGAWATT SM RESEARCH REACTOR

S. M. Feinberg, S. T. Konobeevskii,
N. A. Dollezhal', I. Ya. Emel'yanov,
V. A. Tsykanov, Yu. M. Bulkin, A. D.
Zhirnov, A. G. Filippov, O. L. Shchipakin,
V. P. Perfil'ev, A. G. Samoilov, and V. I.
Ageenkov

Translated from *Atomnaya Energiya*, Vol. 8, No. 6,
pp. 493-504, June, 1960

Original article submitted March 15, 1960

The present article describes the design of the Soviet SM research reactor, where the neutron flux attains $2.2 \cdot 10^{15}$ neutron/cm²·sec. The intensive neutron and γ -ray fluxes that are obtained in the reactor make it possible to perform many investigations in the fields of nuclear physics and reactor techniques.

The SM reactor operates with intermediate neutrons and is characterized by a high ratio of the maximum neutron flux to the heat output. This ratio determines the efficiency of any research reactor. For the SM reactor this ratio is equal to $4.4 \cdot 10^{10}$ neutron/cm²·sec·kw.

The installation of this reactor represents a considerable advance in the development of Soviet reactor construction. The experience gained in the design and utilization of Soviet research reactors was used to a great extent in constructing this reactor.

This article is not concerned with a description of the physical characteristics of the reactor; the main emphasis is laid on the description of the engineering solutions on which the reactor design is based.

Purpose of the Reactor

The SM reactor is intended for scientific-research investigations connected with the use of the intensive fluxes of thermal and fast neutrons and γ -rays which are obtained in the reactor. The reactor operates with intermediate neutrons and has a sufficient reactivity margin for simultaneous investigations in all experimental channels as well as in beams. In correspondence with a planned program of investigations in the SM reactor, the following experiments will be performed first:

- 1) production of new transuranic elements;
- 2) a study of the properties of fissionable and structural materials in neutron and γ -ray fluxes at different temperatures (from 20° K to 2000° C) and in different media (gas, water and pressures from 50 to 350 atm, liquid metals, etc.);
- 3) investigations of the spectrum of intermediate neutrons by means of the spectrometry method;
- 4) investigations of the spectrum of γ -rays from the (n, γ) reaction;
- 5) a study of short half-life radioactive isotopes;

6) investigations pertaining to neutron diffraction photography.

The reactor design also makes it possible to perform other investigations.

Basic Characteristics of the Reactor

The efficiency of experimental reactors depends to a considerable extent on the ratio of the maximum neutron flux in the reactor to its heat output. The larger this ratio, the smaller the fuel expenditure. Therefore, in designing the SM reactor, efforts were made to secure the maximum value of this ratio. Investigations showed that, in intermediate reactors, this ratio can be considerably greater than in thermal reactors*.

The core of intermediate reactors has a smaller volume; moreover, for these cores, one can use structural materials which permit the attainment of high energy intensities. Since the density of neutrons which are slowed down in the reactor central water volume and in the lateral reflector is proportional to the energy intensity, such reactors make it possible to obtain a great thermal neutron density for a comparatively low power level.

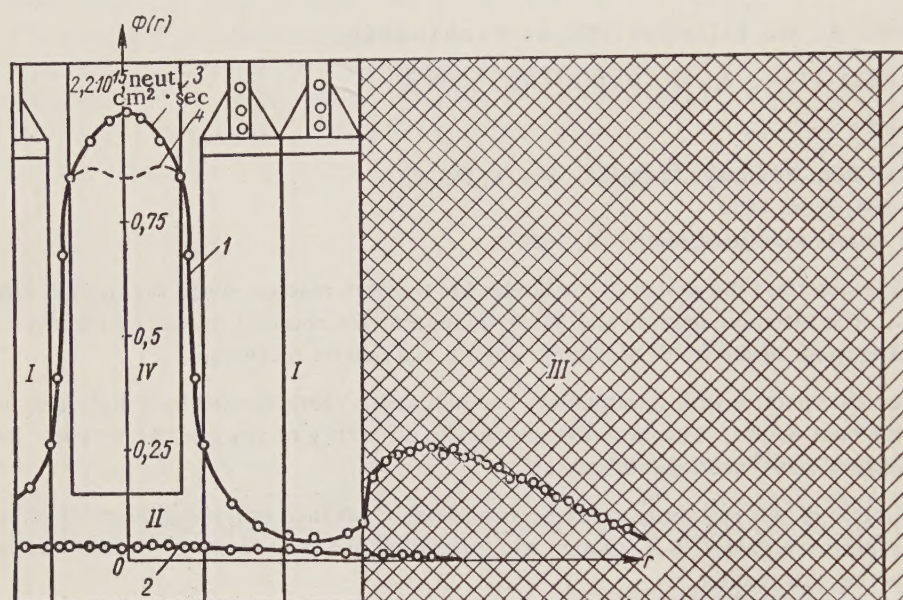


Fig. 1. Neutron flux distribution along the reactor radius. I) Core; II) central water volume; III) reflector; IV) central experimental channel; 1) U^{235} fission density measured by means of a fission chamber; 2) U^{235} fission density measured by means of a fission chamber encased in cadmium; 3) without a specimen in the channel; 4) with a specimen in the channel; the specimen absorbing power (with respect to thermal neutrons) corresponds to 15 g of U^{235} .

Thus, in the SM reactor, for a power of 50 Mw, the maximum flux of thermal neutrons in the lateral reflectors attains $5 \cdot 10^{14}$ neutrons/cm²·sec, and, in the central water volume, this flux attains $2.2 \cdot 10^{15}$ neutron/cm²·sec. Thus, the ratio of the maximum thermal neutron flux to the power level in the SM reactor is equal to $4.4 \cdot 10^{10}$ neutron/cm²·sec·kw. In certain thermal reactors, this value is considerably lower. The neutron flux distribution along the SM reactor radius is shown in Fig. 1.

Another important advantage of intermediate reactors is their longer period of operation. Thus, for a U^{235} burn-up depth of 25%, the SM reactor can operate without recharging over a period of 60 to 65 days. However, from the physical point of view, it is advisable to replace some of the individual burned-up slugs with fresh ones. In this case, the U^{235} content in the core will change very little during operation and the rate of the fuel burn-up will be maintained at the approximately average level, which is equal to 12.5%. At the same time, the duration of

*S. M. Feinberg, et al., Proceedings of the Second International Conference on the Peaceful Uses of Atomic Energy (Geneva, 1958). Reports by Soviet scientists: Nuclear Reactors and Nuclear Power Engineering [in Russian] (Atomizdat, Moscow, 1959) Vol. 2, p. 334.

the reactor run without recharging is determined by the capacity of the box with spare slugs, which is placed inside the reactor. The capacity of the SM reactor box secures a continuous operation over a period of 40 days. Moreover, an intermediate reactor is much less subject to contamination by xenon, and it makes it possible to produce neutron fluxes with different energy spectra.

The reactor core consists of slugs with fuel element plates containing 90%-enriched uranium. The lateral reflector is made of beryllium oxide. The lateral reflector size was determined by taking into account the necessary arrangement of all experimental devices in sufficiently large fluxes of thermal and fast neutrons. The total volume of the lateral reflector is 485 liters, while the volume of the experimental devices amounts to 43 liters, and the volume of the cooling water flowing through the slits between the beryllium oxide bricks is 17,5 liters.

The critical mass of the reactor without the experimental channels is 7.3 kg of U^{235} , and the critical mass when the experimental channels are taken into account is 9.5 kg of U^{235} . The reactor charge for the medium burn-up rate, when the core consists of 20 slugs with different burn-up depths, is ~ 11.7 kg. For a nominal power of 50 Mw, the average energy intensity is 2100 kw/liter, and the maximum thermal flux emitted from the surface of the fuel elements is equal to $5.5 \cdot 10^6$ kcal/m²·hr.

In order to secure reliable heat dissipation for such large thermal fluxes, it was necessary to raise the pressure in the reactor vessel to 50 atm. The cooling water velocity in the core is equal to ~ 9 m/sec for an over-all coolant discharge of 2000 t/hr in the first loop. Under these conditions, the maximum temperature of the fuel element surface does not exceed 195°C.

The Reactor Experimental Channels

For experimental purposes, the SM reactor is provided with five horizontal, one inclined, and 15 vertical channels.

The horizontal channels are located in the plane of the central reactor core cross section (Fig. 2 and 3). The channels are extended to the physical measurement room. Each channel is provided with a protective device, which makes it possible to make preparations for the experiments while the reactor is in operation. The inside space of the channels, including the protective devices, can be evacuated if necessary. The neutron flux at the channel exits is equal to $\sim 3 \cdot 10^{10}$ neutron/cm²·sec.

With the exception of the central experimental channel, all the vertical channels are located in the reflector at different distances from the core center.

The following vertical channels are installed first.

- 1) Three channels for the production of transuranic elements, one of which (with a diameter of 90 mm) is placed at the core center, and the other two are placed in the reflector. The channels are cooled by water which is under a pressure of 50 atm and whose temperature is 60°C. However, the technological layout provides the possibility for the channel operation at higher temperatures and pressures.
- 2) Two low-temperature channels with a water coolant for metallographic investigations. The channels have a special cooling loop with water at temperatures from 30 to 80°C and under pressures of up to 50 atm.
- 3) Two high-temperature channels with a water coolant for the testing of fuel element specimens, investigation of water chemistry problems, and for studies of the corrosion of structural materials. The channels have a special cooling loop with the following water parameters: temperature at the channel upstream end: up to 400°C; pressure: up to 350 atm; water discharge through a single channel: 30 t/hr.
- 4) Five channels with a gaseous coolant for studying the behavior of fissionable and structural materials in fluxes of fast and thermal neutrons and γ -rays in the temperature range from 0 to 600°C. In order to improve the removal of heat from the specimens and to eliminate the effect of the medium, high-purity helium is used as the coolant. All the channels are combined into three independent loops, which makes it possible to perform simultaneous experiments under three different regimes. The helium pressure in the loops is from 30 to 50 atm, and the gas discharge rate in each channel attains 350 kg/hr. The channel design makes it possible to unload the specimens under investigation without shutting down the reactor. Each of the channels is provided with hermetically sealed electrical lead-outs for connection to measuring instruments.

5) A channel cooled by gaseous helium or liquid hydrogen, which is used for studying the behavior of materials at low temperatures.

6) A channel with a gaseous coolant for the testing of specimens at temperatures of the order of 2000°C .

7) A channel with a liquid-metal coolant whose temperature is approximately equal to 1000°C , which is used for studying the behavior of coolants and testing fuel element specimens.

Reactor Design

From among the specific requirements which mainly determined the reactor design, the following should be mentioned:

1) provision of a small-size core which would be capable of withstanding very large thermal loads over a long period of time;

2) provision of adequate core cooling for the above conditions;

3) placement of a maximum number of experimental channels in the immediate vicinity of the core as well as in the reflector;

4) provision of facilities for unloading the slugs without lowering the pressure in the system.

In designing the reactor, special attention was paid to the calculation of the carrying structures in the immediate vicinity of the core, which, while subjected to external loads (proper weight, weight of the mounted parts, and pressure drops), simultaneously experience considerable thermal stresses due to the internal radiation heat release.

The general reactor layout is shown in Figs. 2-5.

The reactor vessel contains the core, the reflector, the protective screens, the control system rods, the mechanism for the internal reloading of slugs, and a number of carrying structures.

The slug cells are located in a square space with a surface area of 420×420 mm, which is separated from the reflector masonry by zirconium sheets. Instead of the four corner cells in the square, guiding zirconium tubes, inside which the emergency protection rods move, were installed. The remaining 32 cells can be occupied by slugs, which are fixed by their stems in the support slab.

During the reactor operation, several of the cells can be occupied by beryllium oxide blocks or remain free. Thus, for instance, if four slugs are withdrawn from the core center, a water-filled space with an area of 140×140 mm is formed, where the central vertical channel can be inserted.

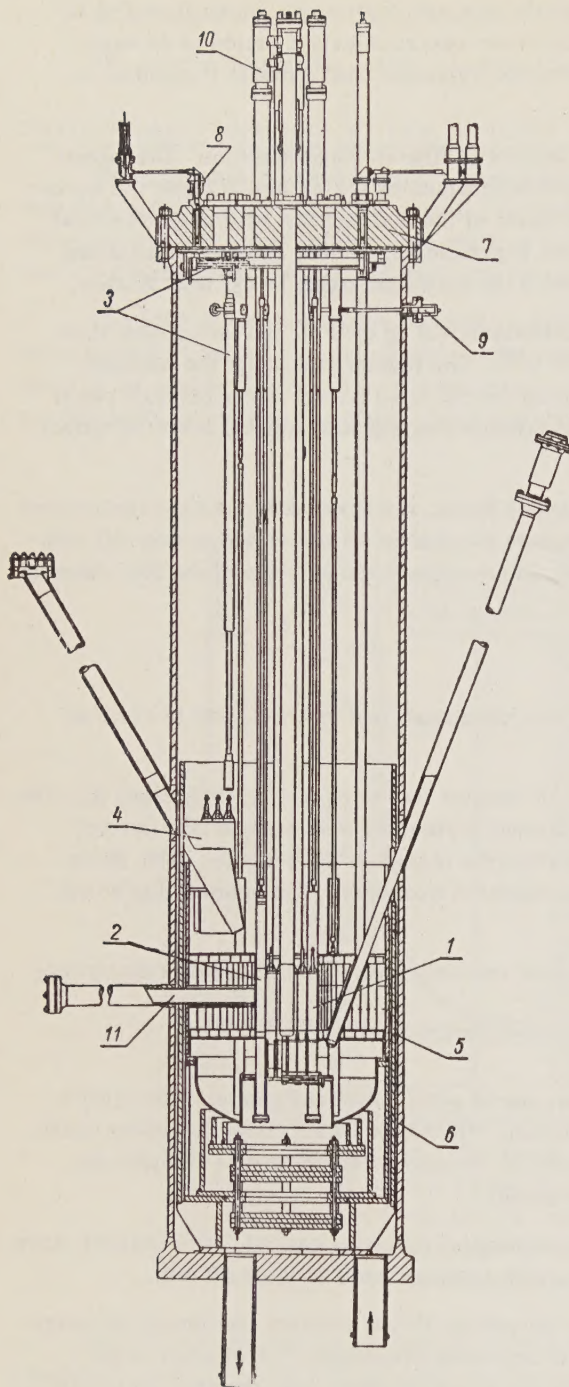


Fig. 2. SM reactor cross section. 1) Core with the reflector; 2) control rods; 3) internal reloading mechanism; 4) reserve cells for slugs; 5) protective screen; 6) reactor vessel; 7) lid; 8) packing-gland lead-out of the reloading mechanism; 9) packing-gland lead-out of the control mechanism; 10) drive of the emergency protection mechanism; 11) horizontal experimental channel.

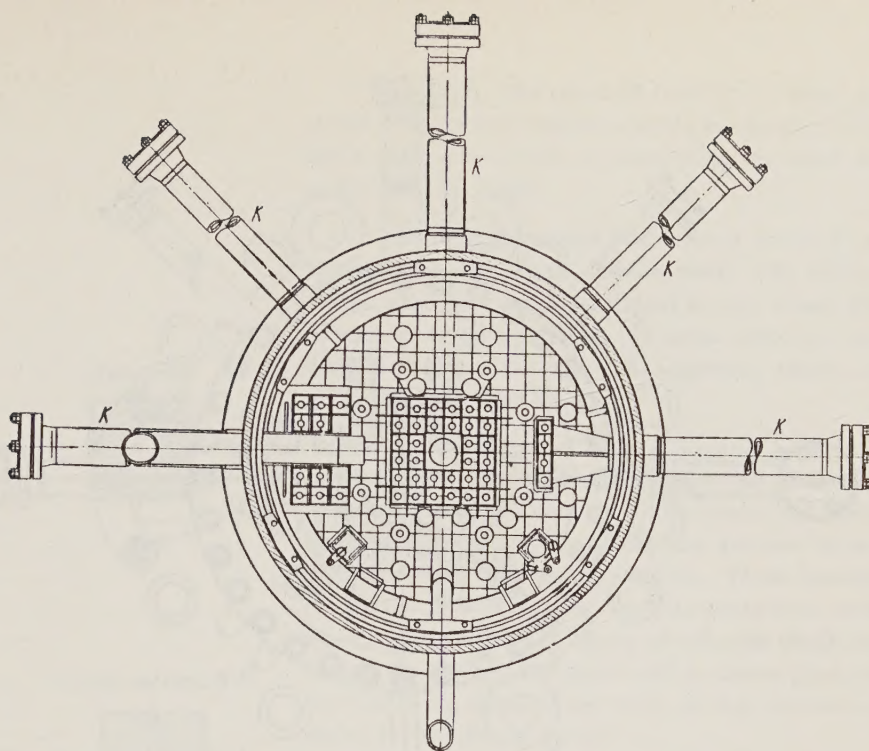


Fig. 3. Top view of the reactor core (K: horizontal channels).

Cells for the storage of fresh slugs and the temporary storage of spent slugs are provided above the reflector. The slugs are transferred by remote handling by means of the internal reloading mechanism without lowering the pressure in the reactor. This mechanism is mounted under the reactor lid and it has two mobile rods, each of which serves one half of the core. After all the spare slugs are used up, the spent slugs are discharged through an inclined channel into the storage space. Two television cameras are provided for observing the reloading in the reactor.

Reactor Vessel and Lid. The reactor vessel (see Fig. 2) is welded; its cylindrical part is made of Stainless Steel 1Kh18N9T sheets, which are 36 mm thick. At the level of the middle core cross section, five horizontal branch pipes for channel mounting are welded to the reactor vessel. Two inclined branch pipes, one of which is intended for the unloading of slugs and the other for mounting the channel with the liquid-metal coolant, are installed at an angle of 90° (plan) above the reflector. Eight branch pipes which serve as the water coolant inlets and outlets are welded to the forged flat bottom. Eight branch pipes which serve as the lead-ins for the drive shafts of the compensating rods and the automatic regulator rods are mounted in the upper cylindrical part of the vessel. Ten nipples, which serve for the outlet of pulse tubes of the system for controlling the activity of the water leaving the slugs, are welded to the same part of the vessel. The vessel lid was made flat in order to facilitate the arrangement of experimental channels on it. The packing between the vessel and the lid consists of a thread-like gasket. Several openings of different diameters were made in the lid for the experimental channels, control rod drives, shafts of the internal reloading mechanism, for the charging with fresh slugs, and for the lead-outs of television camera cables. The unused openings are closed by means of plugs.

The wall thickness of the vessel shell was calculated with respect to the portion in the vicinity of the core, where the vessel strength is weakened by the five welded horizontal channel nipples and where the thermal stresses are at the maximum. Moreover, the spots at the joint between the cylindrical portion and the bottom with the flange were investigated on an optical model. These investigations made it possible to find the optimum joint shape.

In making the vessel, special attention was paid to securing a coaxial position of the horizontal branch pipes with respect to the openings in the screens and the bushings in the reflector as well as a coaxial position of the cells in the reflector with respect to the openings in the lid. In order to satisfy these requirements, a special arrangement simulating the reactor inside space was used at the plant where the reactor was built.

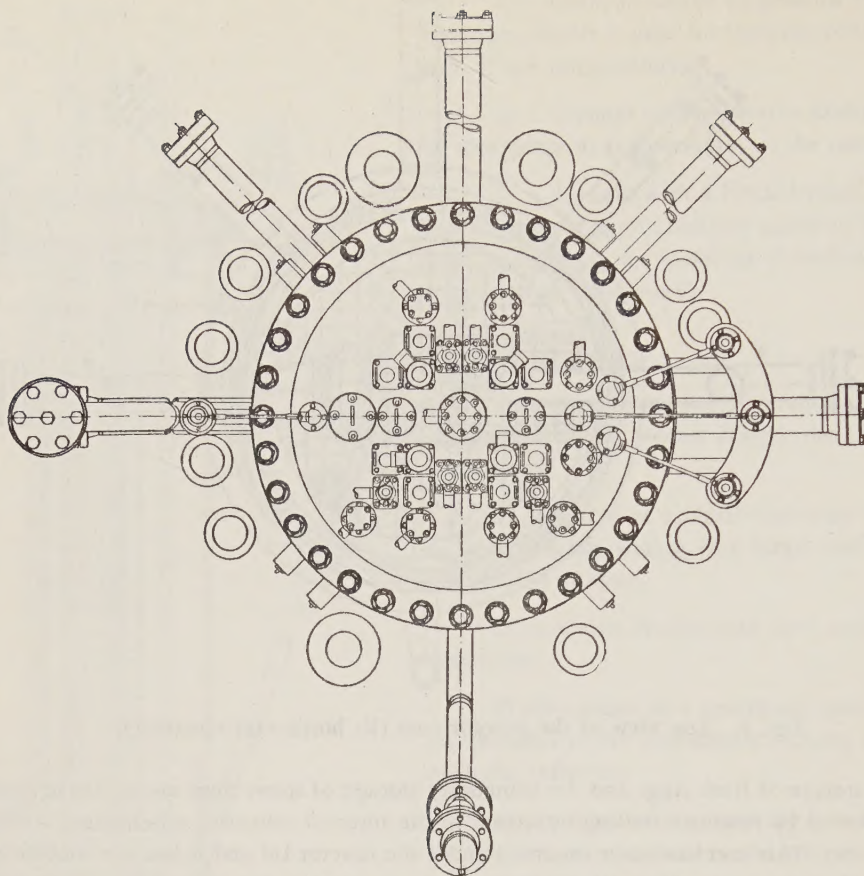


Fig. 4. Top view of the reactor lid.

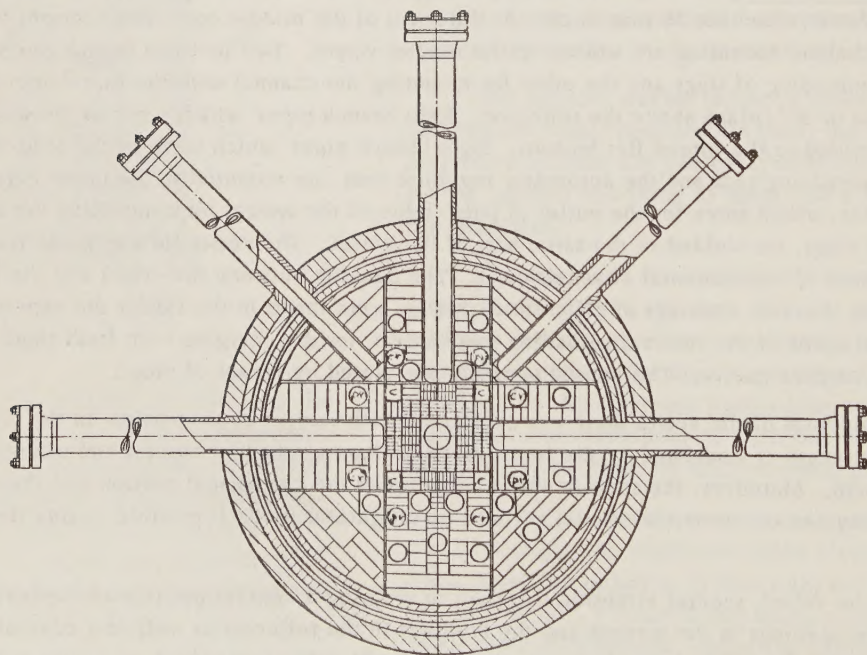


Fig. 5. Reactor section through the horizontal channels.

Reflector. The reflector (see Fig. 2) forms a structural unit which includes the beryllium oxide brickwork, a slug holder plate, and a pulse tube system for controlling the water activity at the outlet from the slugs.

The beryllium oxide brickwork is enclosed in a shell, which is made of stainless steel 25 mm thick. The reflector shell serves at the same time as an additional reactor vessel shield. In order to secure compact packing, the entire reflector brickwork was divided (in the plan) into eight segments, which were separated by means of zirconium sheets.

A large amount of heat is liberated in the reactor reflector ($\sim 40 \text{ w/cm}^3$ in the first layers) due to the absorption of γ -radiation emanating from the core. For this reason, the dimensions of each block were determined by taking into account the maximum allowable thermal stresses in these blocks. These conditions as well as the presence of 24 vertical and five horizontal channels in the reflector led to a great diversity of reflector block shapes (approximately 65 types). The upper and the lower grids between which the blocks are installed are made of thin stainless steel sheets, which are reinforced by ribs.

The slug holder plate consists of 32 cylindrical bushings, which are assembled between two sheets. The bushings, the spacing between which is equal to 70 mm, have a bored collar with exact tolerances at the lower end and a conical outlet at the upper end. The positive placement of slugs is secured by the accuracy in making the slug stems and the holder plate, which is fixed on a cylindrical bushing, which is welded to the bottom of the reflector shell. The system of pulse tubes for controlling the water activity is fixed on the holder plate from below.

In order to reduce the number of pulse tubes which are extended from the reactor vessel to the data transmitters, the coordinate system of water sampling was used. The operating principle of this system consists in sampling water from each row of slugs into collectors, the number of which corresponds to the number of rows. Another row of collectors, whose arrangement is similar to that of the first row, is installed perpendicularly to the first row. Thus, if a fuel plate is damaged, the water with higher activity enters two mutually perpendicular collectors; the damaged slug is directly determined with respect to the designation numbers of these two collectors.

Slugs (Fig. 6). The fuel element type and the design of the jacket for the fuel elements were chosen with the aim of securing a compact core with the greatest possible heat exchange surface.

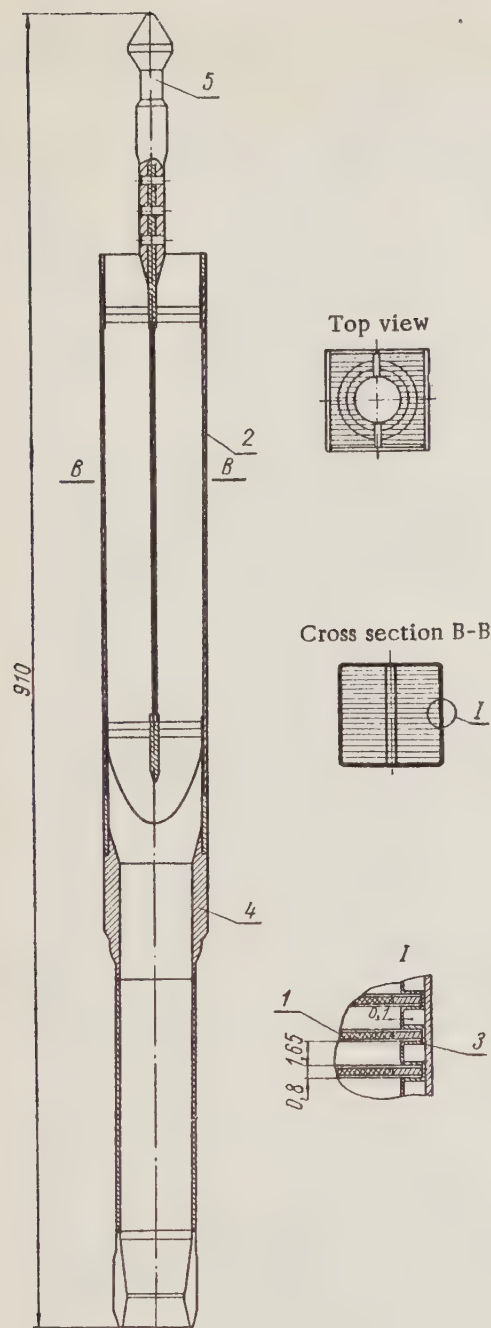


Fig. 6. Slug with laminated fuel elements. 1) Fuel element; 2) slug jacket; 3) spacing rack; 4) stem; 5) holding head.

The fuel element (Fig. 7) is of laminar shape and is prepared according to the following method:

- 1) The core is pressed from uranium oxide powder and electrolytic nickel;
- 2) The core is inserted in the nickel frame and is enclosed on both sides by nickel sheets which are welded to the frame along the entire contour;
- 3) The frame and the core are hot-rolled in an inert-gas medium until the assigned dimensions are obtained; in rolling, the nickel in the core is diffusion-welded to the covering sheets and the frame.

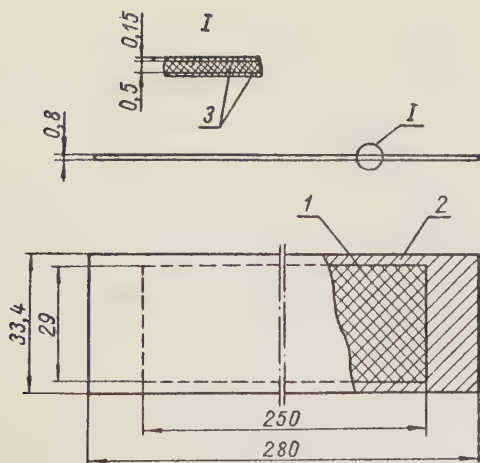


Fig. 7. Fuel element. 1) Fuel (core);
2) frame; 3) covering.

The design of the fuel element and the technology used in its manufacture provide a reliable thermal contact between the heat exchange surface and the granular uranium oxide.

After rolling and cutting, the final dimensions of a plate are the following:

Thickness, mm	0.8
Length, mm	280
Width, mm	33.4
Core thickness, mm	0.5
Core length, mm	250
Core width, mm	29
Over-all plate weight, g.	60
U ²³⁵ weight, g.	12.5

made of stainless strip steel 0.1 mm thick along the entire length. The slug jacket (see Fig. 6) is made of two halves, which are connected by spotwelding. Each half is made of a stainless steel sheet 0.5 mm thick. At the upper end, the jacket is welded to the rack, which is provided with a holding head, and, at the lower end, the jacket is welded to a cylindrical stem with gradual transition to a square cross section.

The plates are assembled in the slug jacket. Each slug contains two rows with 54 plates in all, which are spaced by means of racks at the top and bottom and by means of corrugated strips

Since the fission density of uranium nuclei at the interface between the active zone and the water-filled space sharply increases, the end plates in some slugs contain a smaller amount of uranium in order to equalize the heat release. Thus, three types of slugs, which differ from each other by their end plates, are placed in the reactor. In order to avoid possible errors in placing them, the slug stems are provided with grooves which are oriented with respect to the plates with a lower uranium content, and the holder plate is provided with pegs which fit the stem grooves.

The correctness of plate mounting, the spacing method, and the jacket rigidity were checked by experiment. At the same time, the plates were tested in the reactor loop under conditions close to the actual operating conditions.

Moreover, the problem of the heat transfer in narrow slit channels was studied, as a result of which the necessary data on the wall temperatures and the critical thermal loads were obtained.

Protective shields. The reactor vessel, which is stressed by the internal pressure, is protected from intensive neutron and γ -ray fluxes by the reactor shields (see Fig. 2).

The thickness of the shields and of the reflector shell was chosen with the aim of securing a sufficient flux of neutrons beyond the reactor vessel for a normal operation of the ionization chambers and of keeping the thermal stresses in the vessel below the maximum allowable value. The shields have a cylindrical shape; the inside cylinder is welded to a support ring, and the outside cylinder is fastened to the lower adapter. The inside shield cylinder has a flange, on which the reflector shell is mounted by means of the support ring. The end of the outside cylinder is located above the core. A receiver for spent slugs, cells for the storage of fresh slugs, and an additional shield, which protects the upper reactor vessel part from radiation, are fastened on the cylinder upper end. Slabs which protect the vessel bottom from radiation are installed on the lower shield support ring.

The shields also serve for dividing the reactor inside volume into two zones. This division is secured by closely fitting the lower portion of the shields to the vessel bottom. As a result of this, two volumes are formed: a ring-shaped volume with the four branch pipes for the water coolant supply and the central volume with the welded four outlet branch pipes.

Water coolant circulation in the reactor. The water enters the ring-shaped clearance formed by the shields and the reactor vessel through four branch pipes, which are welded to the reactor bottom. In the ring-shaped clearance, the water supply is divided: one portion is directed under the reflector, where it cools the beryllium oxide blocks, and the other portion passes between the vessel and the shields. At the top, both water streams meet, and the water flows downwards between the slugs, through the clearances between the demountable beryllium blocks,

and along the emergency protection rods. The water then passes through the clearances in the lower protective slab and comes out of the reactor through four central branch pipes, which are welded to the vessel bottom.

Since a certain given amount of water must be transmitted through the reflector, adjusting baffles, by means of which the water flow is regulated during preliminary adjustments, are fastened to the upper portions of the shields. During the adjustment operations, the water flow is controlled by means of a flowmeter, which is connected to two pulse tubes, one of which is installed upstream from the reflector water coolant inlet and the other is installed downstream from the outlet.

The presence of fitting spots between the water coolant inlet and outlet creates the possibility of water leakages. In order to minimize the leakages, all the fitting spots (between the shields in the vessel, between the reflector shell and the shields, and the spots where the slugs are placed) are carefully adjusted and individually checked after final assembly.

The Control System

The reactor control system includes two automatic regulators with coupled rods, four compensating rods, and four emergency protection rods, which can also be used as compensating rods.

Thirteen ionization chambers, which are placed outside the reactor vessel, serve as the control system data transmitters.

The automatic control system maintains the assigned reactor power level within the 0.5-100% range.

A rack and pinion drive is used as the final mechanism of the automatic regulator rods. From the servomotor and the reduction box, which are mounted on a platform above the reactor, the rotation is transmitted by two shafts with universal joints to the two pinions, which drive the racks with the rods. Packing glands with an arrangement for collecting the leakage flow and directing it into a hermetically sealed draining system are provided at the spot where the shafts enter the reactor. The maximum speed of the rods is 40 mm/sec, and the rod stroke is 450 mm.

The emergency protection system consists of three independent channels in the electrical circuit, and it operates if the power level exceeds the nominal value by 25%. Moreover, the emergency protection system comprises a preventive protection arrangement, which is activated if the power level, which is maintained by the power control device, is exceeded by 10-15%. On the reception of this signal, the automatic regulator rods are introduced into the core. The action of the preventive protection system stops if the power level drops by 2-7%.

The emergency protection servomechanisms are designed to secure a quick trip-out of the rods for any position they may occupy. This provides positive emergency protection, and, at the same time, makes it possible to use the rods for a partial compensation of the reactor reactivity. The emergency protection drive comprises a motor which rotates at low speed. The motor rotor is rigidly connected to a ball nut. A screw with a core made of magnetic stainless steel at its end moves inside the nut. The lower motor flange is fastened to the frame of the power and signal coil system. A magnetic core, which is connected to a pole, at the end of which the rod is suspended, is located inside this system. When the power coil system is energized, the resulting cohesive force between the cores vanishes, and the rod falls into the core under the action of its own weight and the pressure drop. At the end of the falling period, the braking device is activated.

The over-all rod stroke is 400 mm, and the reactivity compensation section is equal to 150 mm.

The data transmitters for the emergency protection signals consist of three ionization chambers, which are surrounded by lead shields.

The design of the final mechanisms of the four compensating rods is similar to that of the automatic regulator rods. The rod speed is 1 mm/sec, and the rod stroke is 450 mm.

Two types of rods of different shapes are used in the reactor. The emergency protection rods have a square cross section, and the others have a round cross section. Each rod has an upper and a lower part. The upper (absorbing) part of the rod consists of a cadmium tube, which is enclosed in a stainless steel jacket. The tube is filled with water. The lower part of the rod is made of beryllium oxide. Thus, in introducing the rod absorbing part into the reflector, beryllium oxide is replaced by the cadmium tube, which is filled with water. The efficiency of this type of rod in a reactor with intermediate neutrons is sufficiently high.

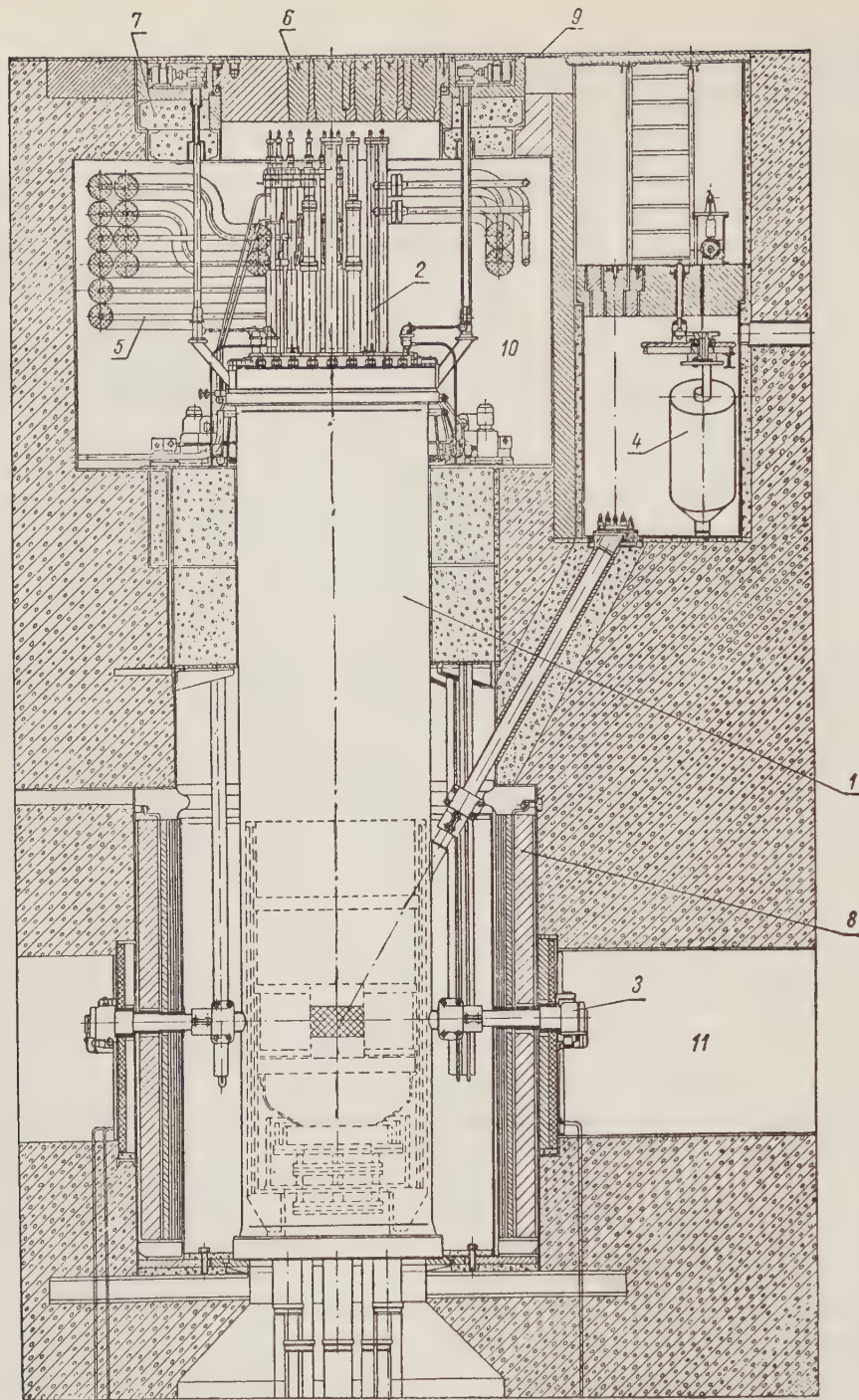


Fig. 8. Reactor position inside the shield. 1) Reactor; 2) vertical experimental channel; 3) horizontal experimental channel; 4) unloading mechanism; 5) piping of the vertical channels; 6) rotary disk slab; 7) top protective slab; 8) thermal shield; 9) reactor room floor; 10) space above the reactor; 11) space for the protective devices of the horizontal channels.

Under start-up as well as operating conditions, the reactor power measuring system makes it possible to measure the power level on a linear and a logarithmic scale, to record the power level, to vary its rise time, and to provide sound signalization. It is considered that the lowest measurable reactor power level is $10^{-5}\%$.

The Reactor Shielding

The reactor location in the shield is shown in Fig. 8. The reactor rests on a support slab inside a concrete pit. A passage, through which the water coolant piping passes, leads to the lower portion of the pit.

In order to secure the maximum intensity of neutron fluxes at the beam exits, the shield near the location of the horizontal channels is made as heavy as possible; it consists of air-cooled steel shields and heavy concrete. The thickness of the protective shields was determined by taking into account the maximum allowable temperature drop in concrete. The air in the space between the reactor vessel and the thermal shield is rarefied to such a degree that the active air does not penetrate into the space above the reactor and into the steel shield cooling system.

The horizontal channels at the outlet from the thermal shield are made airtight, whereby the space where the protective devices are located is separated from the reactor space.

The top of the reactor emerges into the room above the reactor (Fig. 8). This space contains the control system drives, the piping of the experimental channels, and the channels of the ionization chambers, which are fastened to the shielding blocks that separate this space from the reactor space. Moreover, the space above the reactor contains cable conduits leading to the electric motors, ionization chambers, leakage detection data transmitters, and the experimental channel data transmitters.

The space above the reactor is separated from the reactor room by means of a shielding slab, and, above the reactor itself, this space is separated from the reactor room by a rotating cast iron disk with several interchangeable plugs, which are located above the experimental channels. The placement and withdrawal of the specimens under investigation is effected by manipulation from the reactor room, while the disk does not have to be rotated for this purpose. The disk is rotated only in the case where it is necessary to withdraw a channel or to perform work on tightening the reactor lid packing. All the equipment not connected with investigation work is contained in the space above the reactor, which makes the experimenters' work safer and more efficient.

Adjacent to the space above the reactor is the liquid-metal loop adjoining basement, the well containing the mechanism for unloading spent slugs, and the storage room for these slugs, which is filled with water. The unloading mechanism is connected to the reactor through an inclined tube, which is covered by means of a demountable lid.

The slugs are withdrawn from the reactor in the following manner: the pressure in the reactor is released, the inclined tube lid is taken off, and the unloading mechanism, which aligns the container with the inclined tube, is activated. The adapter for gripping the cases is then lowered; the adapter pulls the slugs out of the reactor and hauls them into the container, after which the container rotates until aligned with the inclined channel connecting the mechanism frame with the storage room, and the slugs are lowered and placed into the storage cells. The entire process of unloading the slugs takes place under a protective layer of water.

The Reactor Cooling Layout

The water temperature at the reactor upstream end is 50°C ; at the downstream end, it is 80°C . The reactor cooling system comprises four independent loops, each of which includes a hermetically sealed circulation pump with a capacity of 500 t/hr, which produces a pressure of 10 atm, a 15-Mw heat exchanger, and a purifying filter. Two of the four pumps are connected to the emergency-feed line, and, in the case of electric power failure, the power for these pumps is supplied from a storage battery. Moreover, the layout comprises a loop for emergency reactor cooling. This loop is designed for maintaining a steady water circulation during the time required for switching the electrical power supply lines of the main pumps to the storage batteries, and it includes a hermetically sealed pump and a heat exchanger. The pump is in constant operation and is fed from a storage battery.

For continuous elimination of radioactive admixtures from the water and for removing detonating mixtures, the layout includes ion exchange filters and a bypass system with contact apparatuses, which are calculated for the recombination of $20 \cdot 10^3$ norm liters of detonating mixture.

The system also contains volume compensators. The pressure in the loop is maintained by means of helium, which is fed to the compensators through a pressure-reducing regulator from a tank system. In releasing the pressure, helium is admitted into the receiving tanks.

NEW IDEAS IN THE STRUCTURAL DESIGN AND LAYOUT OF NUCLEAR REACTORS

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This article discusses methods used in siting reactor buildings, and their structural layouts. The advantages and the disadvantages of the above-ground and below-ground approaches in the construction of reactor buildings are considered. Current techniques for building concrete shielding around stationary power reactors are described. Detailed treatment is given to problems in the use of explosion-proof and missile-proof protective shells, and problems concerning the economics of the use of various compositions of concrete for biological shielding of reactors are elucidated.

Reactor Siting Methods

Siting of reactors below ground level has been a common approach to reactor siting up to the present. As some examples of such comparatively costly and laborious installations, we may cite the reactors of the World's First Nuclear Power Station (USSR) and the Shippingport nuclear power station (USA). This solution is owed to the effort to secure greater safety for the population of the surrounding areas in case of damage to the reactor. Underground excavation of reactor sites has been practiced primarily when building new reactors of untested design, and especially in cases where pressurized water was employed as coolant.

At present, most reactors are being built above-ground, with the advantages consisting, as a rule, of greater simplicity in design, lower construction costs, and shortened construction time, resulting from elimination of the need to excavate deep foundation pits or build foundation walls designed to withstand heavy earth pressure, not to mention being leaktight against passage of ground water. According to the data reported by Bergstrom and Chittenden, in reference to the design of the experimental power reactor EBWR, it was found that excavation costs increase the total cost of the reactor installation by as much as 7% [1]. Inspection and overhaul of the reactor is much simpler when the reactor is built above-grade. It should also be noted that in the case of reactor damage and attendant leakage of radioactive fluids, there is a great hazard of these contaminants getting into the surrounding earth in the case of a reactor recessed below ground level. The above-grade position of research reactors facilitates convenient placing of all the experimental equipment servicing the reactor. The above-grade variant is mandatory for horizontal reactors, inasmuch as removal of fuel elements from a below-ground horizontal reactor would require an underground reactor hall of enormous size.

The present shying away from underground installations is also explained by the increased reliability of the control systems of reactors, and likewise by the experience accumulated in operation of reactors of many types.

It is very seldom that reactors are built completely below ground level, since this would entail greatly increased costs and much more troublesome operating conditions. Below-grade siting is motivated primarily by a striving to secure airtight shielding of the object. However, in Scandinavian countries and in Switzerland, with their great knowledge of tunnel and excavation engineering, construction of underground reactor facilities is still being planned and carried out. Examples representative of such reactors are the Norwegian heavy water reactor at Halden [2, 3], and the Swedish power reactors R-1, R-3, "Adam", and "Eva" (Fig. 1).

A unique approach to exploitation of the advantages in underground siting is to be noted in the draft project for a large-scale power reactor [4] drawn up by Swedish engineers. The steel pressure vessel of the reactor, capable of withstanding an internal pressure greater than 10 atm, and having relatively thin walls, is housed inside a below-grade pressure-tight chamber such that the pressure inside the pressure vessel is transmitted through the gas flooding the space between the walls of the pressure vessel and the chamber enclosure to the surrounding rock. The walls of the pressure vessel actually play the role of an intermediary membrane in this setup (Fig. 2).

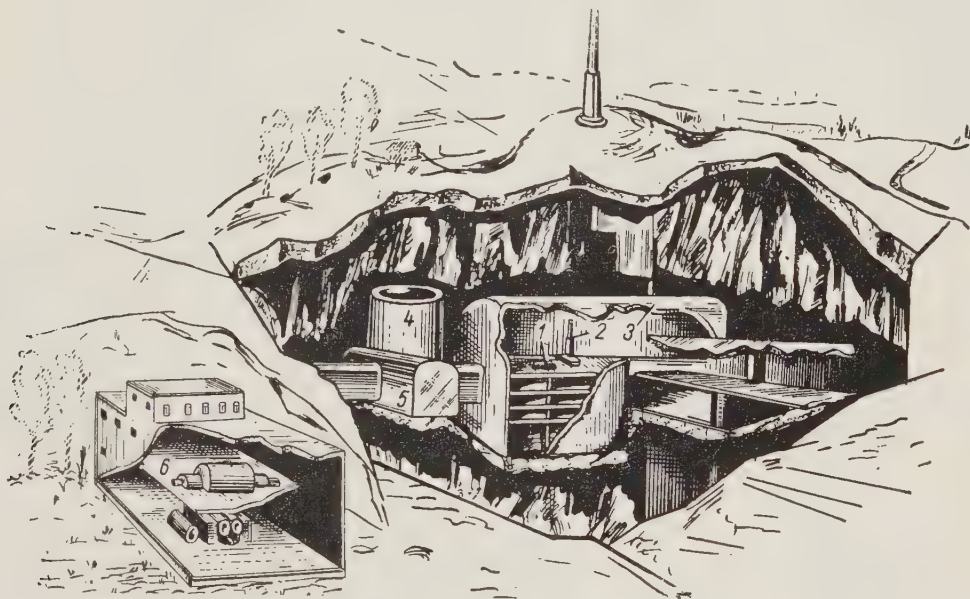


Fig. 1. "Eva" underground reactor installation (Sweden): 1) Heat-exchanger shell; 2) fuel-element charge-discharge; 3) fuel-element storage house; 4) expansion tank; 5) tunnel connecting reactor building to turbine house; 6) turbine house.

The Swedish reactor designers drew the following conclusions from their experience in planning and building underground reactor facilities [4].

The advantages of such reactors are:

- 1) the installation built below-grade will be capable of withstanding excess internal pressure concurrent with reactor damage, without any additional complex safety measures;
- 2) these facilities can be reliably protected against wartime destruction;
- 3) the surrounding rock body provides excellent biological shielding;
- 4) leakage of radioactive gases out into the surrounding medium in the event of damage is significantly diminished by the filtering action of the overlying rock and soil layers, and in consequence of the decay of radioisotopes during their traversal of the walls;
- 5) the underground siting is to be preferred at locations where it is important to preserve the natural landscape intact.

The disadvantages inherent in underground reactor installations are:

- 1) the need for an appropriate rock cover, which narrows down the available freedom of choice for the site;
- 2) the need for careful inspection of ground-water level and ground-water movement, with resulting additional expenses;
- 3) inconveniences due to the limited size of the installation.

Liljeblad and Madsen [4] note that reduction of costs in below-grade excavation work in Sweden in recent years has had the effect of appreciably offsetting the more protracted construction schedule required for underground construction, compared to above-grade work. According to them [4], a tunnel 20-50 m² in cross section and 100 m in length can be driven in a month and a half; a rock chamber 20,000 m³ in volume can be blasted out and excavated in 7 to 10 months. Swiss specialists submit pretty much the same considerations in their arguments favoring underground nuclear electric power generating stations.

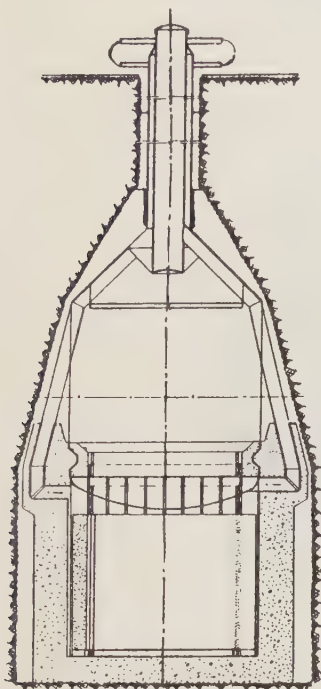


Fig. 2. Cross-section diagram; Swedish plan for an underground power reactor.

On the basis of the experience accumulated in underground construction of reactors in Sweden, it is felt that compartments of the order of 25 m may be safely excavated in rock where the roof thickness is 40-50 m. For roof thickness in excess of 50 m over chambers of cylindrical cross section, compartments 35-40 m can be safely tolerated. Important attention is given to providing watertight concrete lining for the tunnels and excavated chambers.

Concrete lining of the walls of underground rock excavations may be achieved by placing concrete or by applying gunite with a sand-cement grout. The advantage to be gained in discrete upright foundation walls for the underground installation is their accessibility from either side, and the possibility of using the voids for additional drainage of ground water. The disadvantage resides in the increased hazard of damage to the walls if the reactor should sustain an injury of an explosive nature. Gunite injection is a lot cheaper than concrete lining. However, this incurs a penalty of much thinner density of lining, not to mention the uneven surface, which is highly undesirable from the standpoint of deactivation operations (particularly after possible damage, with radioactivity release, has occurred).

It is considered that no concrete lining of any kind is capable of being completely gas-tight. In the Swedish reactors, a concrete-lined chamber with 5000 m² surface area was found to leak 0.01-0.02 m³/sec air at a pressure of 1 atm. If specifications for a gas-tight lining are very exacting, a lining of sheet steel or plastic will be required.

According to Swedish data, at roof thickness of 20 m and an excavated span of 20 m, rock slides may occur at an overpressure of 15-20 atm, which is 5-6 times greater than the usual overpressure which is designed for.

It is the prevailing opinion in the USSR that primarily because of cost considerations and the compressed construction schedules, recourse may be had to underground siting of reactors only where already existing excavations are available (e. g., excavations completed previously for some other purpose, but not now in use), or else mining pits or natural caverns may be utilized.

Some Problems in the Construction of Reactor Enclosures.

Two-echelon reinforced concrete shielding has been resorted to in some of the recent reactor power installations. The inner shielding ring contacts directly with the reactor proper; the outer echelon encases the pipes and conduits leading to the steam generating units. This was the decision taken, e. g., in the design of the British nuclear power stations at Hunterston and Hinkley Point (Fig. 3). In individual cases, the outer echelon of the shielding also reaches to the steam generators (e. g., in the U S "Yankee" nuclear power station, see Fig. 4).

In double-echelon shielding, the thickness of the outer echelon is usually about 60% of the thickness of the inner shielding. This facilitates overhaul of the radioactive piping and steam raising units (when they are shut off) without shutting down the reactor. When the heat exchanger units are separated by baffles, the possibility of closing off and overhauling the individual heat exchanger units without having to interrupt the operation of the reactor is secured.

The lateral concrete biological shielding of reactors, heat exchangers, and the shielding walls of the reactor buildings are placed in monolithic form, as a rule. An exception is presented only by some small experimental reactors where the latter and top shielding is in the form of a module of removable concrete blocks.

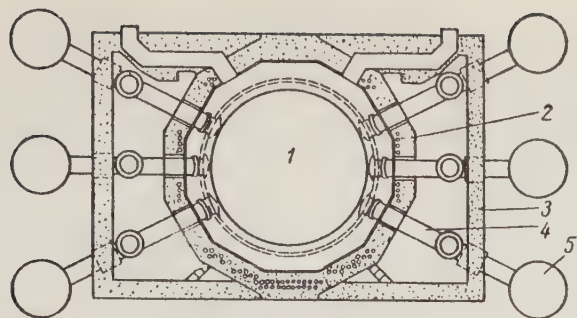


Fig. 3. An example of double-echelon shielding in the reactor building of the Hinkley Point nuclear power plant: 1) reactor; 2) inner ring of biological shielding; 3) outer ring of biological shielding; 4) gas duct; 5) steam generating unit.

The striving to avoid such sectionalized concrete-block shielding, or combinations of sectionalized and monolithic shielding (i. e., with subsequent concreting of the arrays of blocks) in the biological shielding of large reactors is to be explained first of all by the danger of crevices forming between the blocks, particularly in the outer extended zones of the shielding walls, which are subjected to heat on one side only.

However, it has by now become established that local hairline cracks in concrete exert no material effect on the efficiency of biological shielding. This accordingly provides justification for making wider use of modular block shielding and combined shielding designs, facilitating in great measure the industrialization and speeding up of reactor construction work. The wall of one face of the building housing an experimental power reactor now being constructed in the USSR will be assembled in modular fashion, and will consist of reinforced-concrete blocks measuring 5.4 by 2.4 by 1.5 m (Fig. 5). The weight of one block is 46.5 tons. The blocks present a broken-line configuration seen in vertical cross section; they are lined with stainless steel on the side facing the reactor core. Combined modular-monolithic constructions of shielding enclosures (e. g., used for nuclear-reactor enclosures) of the type appearing in Fig. 6 have been gaining favor in the building of more recent reactor installations in the USSR. Large constructions in the form of assembled reinforced-concrete columns, girders, and slabs using standardized prefabricated modular components to maximum advantage are employed in the building housing the experimental power reactor, as well as the building housing the reactor of the Novo-Voronezh nuclear electric power generating station. Such construction designs obviate any need for timber, scaffolding, formwork for placing concrete, etc., and greatly facilitate industrialization and speed-up of a construction work on enclosures.

While on the subject of interchangeable reactor shielding components, we might mention that the system of constructing the top shielding of several layers of reinforced-concrete slabs or blocks with layers of other materials (lead, etc.) sandwiched in between, popular in the past, is more recently being crowded out by the tendency, fully justified by operating conditions, of arranging the top shielding in the form of detachable large "plugs," comprising a metal sheathing encasing a concrete "meat." The weight of each of the constructions amounts to 20 tons and higher.

Applications for Various Concrete Mixes

Data [5] proving the unfeasibility of employing special heavy concretes as biological shielding, except for those cases where it is absolutely imperative to keep shielding thickness within bounds. Others [6-8] assert that the use of metal aggregates and concrete of weight by volume exceeding 4.8 tons/m^3 should be restricted to those cases where the high cost of the concrete is offset by the total savings achieved as a result of minimizing shielding thickness. Horton [9] notes that a concrete shielding arrangement using aggregates drawn from local quarries is almost always cheaper than using heavier coarse aggregates transported from afar. The curves seen in Fig. 7 demonstrate that (with the exception of small reactors) the use of conventional concrete optimizes costs.

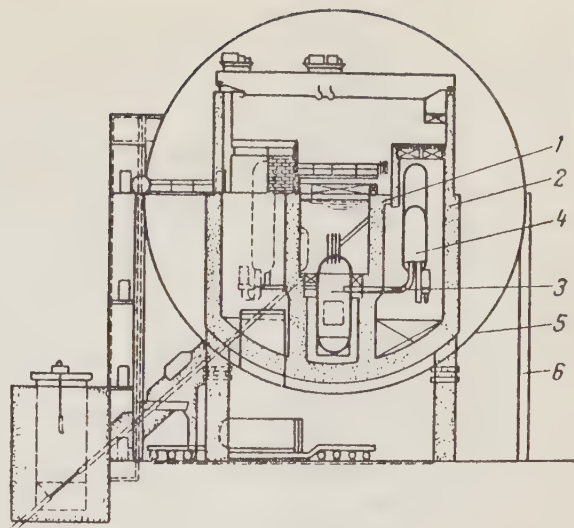


Fig. 4. An example where the steam generators are located inside the outer shielding echelon ("Yankee" nuclear power plant): 1) inner shielding echelon; 2) outer echelon; 3) reactor; 4) steam generator; 5) containment enclosure; 6) support column.

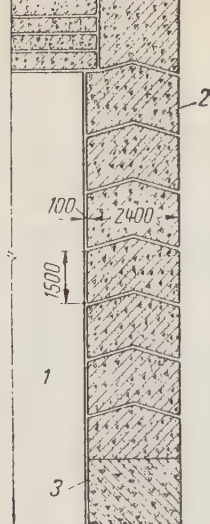


Fig. 5. Modular design biological shielding for experimental power reactor under construction in the USSR: 1) space for reactor core; 2) concrete blocks; 3) stainless steel lining.

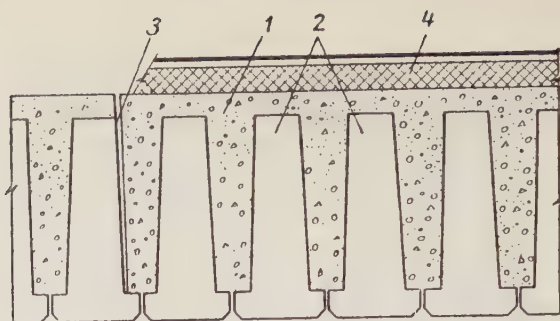


Fig. 6. Layout of the shielding enclosure (cross section through the girders): 1) monolithic concrete; 2) supporting layer of prefabricated reinforced-concrete beams; 3) temperature-expansion seam; 4) top layer.

As for applications in reactors enclosed within steel containment shells, a paper [10] based on the experience of the USA's Stone and Webster Engineering outfit, engaged in the design and construction of shielding enclosures for nuclear reactors, mentions that the cases considered by this firm involving the use of high-density concrete as biological shield with a consequent reduction in shielding dimensions brought about economy in steel costs which more than covered the additional outlays for the high-density concrete. However, there are no published economic calculations or data available on this question at present, and the significant difference in costs of conventional as against special heavy concretes in the USSR compared to prices prevailing in the USA and the UK give us reason to doubt that this view will retain its validity for conditions in the USSR as well.

Until recently, concrete with limonite ore (brown hematite) having a high content of chemically bound water, higher than that characteristic of conventional magnetite and hematite ores (10-17% as against 1-4%), as aggregate has been used in some particular cases for biological shielding. The idea was to bring about a substantial increase in the efficiency of shielding against neutrons. Calculations performed at the Institute of Atomic Energy have shown that to get the same shielding efficiency against neutrons from the use of conventional concrete, the shield thickness would have to be 20% greater than when limonite concrete is employed.

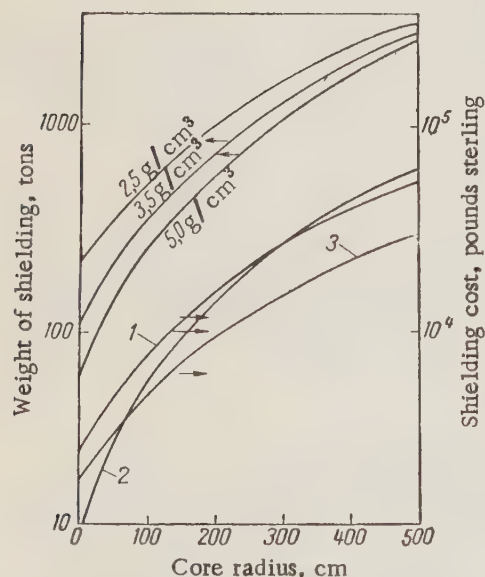


Fig. 7. Weights and costs of a representative set of shielding concretes (data from [8]): 1) barytes concrete (3.054 g/cm^3); 2) concrete with pig-iron punchings (5.3 g/cm^3); 3) conventional concrete (2.5 g/cm^3).

One of the designing institutes in the USSR carried out, for the purpose of a feasibility evaluation of the use of limonite concrete as against ordinary concretes as biological shielding, a projection and comparison of estimated construction costs for a large-scale reactor installation using shielding of ordinary concrete and using the variant of limonite concrete shielding. The comparison was carried out with reference to different regions of the USSR. The actual prevailing conditions in the production of limonite ore, transportation of the ore by railway to the construction site, and practical costs of locally procured concrete aggregates were all taken into account in the estimate. The calculations showed that the use of limonite concrete for shielding in large reactors results in the following cost increases in shielding for the reactor buildings (percentagewise):

For centrally located regions of the USSR	by 12%
For the mid-Ural belt	by 15
For Western Siberia	by 128
For Eastern Siberia	by 132

This comparative estimate permits the inference that from an engineering and costs standpoint, the use of limonite concrete for biological shielding of reactors in the USSR will result in over-all higher construction costs. The use of concretes incorporating limonite or hydrohematite ore aggregates might be justified only in case the reactor is being built in the immediate vicinity of the location where those ores are mined.

Compacted dried iron ore is worthy of mention as a low-cost material for horizontal biological shielding placed in reinforced-concrete laminates limited to the thickness specified in the design. In the shielding assembly of one of the buildings for the proton synchrotron (synchrophasotron) of the Joint Institute for Nuclear Research at Dubna, consolidated dry magnetite ore from Krivoi Rog, with a dry volume by weight 2.62 tons/m^2 at a thickness of 0.9 m for the shielding layer and a total volume of 500 m^3 was used. The ore composition included 85% magnetite crushed rock and 15% magnetite sands. The components of the shielding were laminated into layers 15 cm thick by a 10-ton road leveler. Less tractable lumps were consolidated by using electrically driven tamping rods weighing 200 kg each. The weight by volume of the consolidated magnetite mix was 3.3-3.5 tons per cubic meter. This resulted in savings of 588 thousand rubles compared to the previously planned shielding assembly of heavy-concrete blocks (weight by volume 3.2 tons/m^3). The shielding cost per cubic meter was cut from 933 rubles, according to the first cost estimate, to 464 rubles. Placing of a layer of consolidated iron ore may thus be found useful in horizontal biological shielding units, not loaded statically, for nuclear reactors.

Use of Protective Shields Against Missiles and Radioactivity Release

Despite the fact that no important accidents of an explosive nature accompanied by the release of radioactive liquids, gases, or fragments have occurred in the actual operation of reactors, most large-scale foreign reactors have protective containment shells, mostly metal structures, incorporated into their design. The building of these containment enclosures, despite the appreciable increase in cost and complexity of the reactor installations involved, is due to the effort to ensure the safety of the area surrounding the reactor plant in the case of a theoretically possible accident involving explosion of the reactor core or a rupture of the lines carrying radioactive and pressurized liquids or vapors.

The problem of the use of protective missileproof containment shells has been the subject of theoretical and design-engineering scrutiny in the USA in the recent period. Experimental work is also being carried out on a comparatively large scale. However, one source reports [10] that a decision to forego building such structures might be justified by the distance from populated centers, by the reliability characteristic of a given reactor type, or by the "mild" operating conditions. The safety of reactor performance would be achieved preferably by improved reliability of the associated equipment, rather than by housing the reactor inside containment shells. It is possible that a reactor type with a large margin of safety may be acknowledged as suitable for use on a wide scale, and that favorable experience in the operation of that type will be accumulated; it would then be possible to make a definitive decision to do without such containment structures. The use of the containment vessels acts as a serious hindrance to achieving low nuclear power costs, and all measures should be taken to cut the costs of the containment enclosures or to circumvent them altogether.

As stated in another report [11], the probability (for each set of 100 reactors) of fatalities resulting from a reactor accident is $2 \cdot 10^{-8}$ in the USA. At the same time, the probability of fatalities in automobile accidents in the USA is $2 \cdot 10^{-4}$.

Several American engineers feel that to avoid excessive costs incurred in building containment enclosures, the strength of the latter should be designed to match only 40% of the maximum possible energy of the conjectured reactor explosion.

One view [12] is that the question of the need to build such containment vessels is still unresolved. And only in the case of a substantial potential hazard of a reactor accident plus absence of any possibility of accurately determining the extent of the possible damage would the additional costs associated with erecting the enclosure be justified.

In the 21st semiannual report of the US Atomic Energy Commission (AEC) [13], the need to build containment shells was justified by the argument that one day following shutdown of a reactor rated at 60 Mw power, the amount of radioactive material present within the reactor is equivalent to 300 tons of radium. An explosion of the reactor accompanied by the ejection of radioactive materials into the surrounding locality could be more dangerous than a nuclear explosion insofar as radioactive contamination of the locality is concerned, although

less dangerous with respect to its mechanical effect than any large-scale explosion which might occur in any other branch of industry.

If the installation cannot be isolated, then gas-tight enclosures (vessels) must be constructed around the reactor. The scale of this secondary shielding for each particular reactor will be determined by the location of the reactor. Laying great stress on the correct design of protective containment shells, the US AEC conducted tests on designs of containment structures at the ballistics research laboratory (Aberdeen, Maryland). Experiments on mockups scaled to one-fourth natural size were staged, using appropriately shaped charges of explosives.

The experiments showed that when models of a radioactive reactor vessel are destroyed with the aid of a corresponding amount of explosives, a shock wave is set up which imposes on the outer containment shell a pressure increasing with very high speed.

Despite the acknowledgment that reactor performance records show excellent safety and that containment shells entail high costs, and even in the absence of any convincing proofs justifying the erection of such containment shells, large-scale nuclear installations are enclosed by primarily metal protective enclosures in a number of countries. Furthermore, recent reactor design practice shows that several large reactors are being built without containment shells (e. g., the British nuclear electric power stations with graphite-moderated gas-cooled reactors, the Soviet nuclear power stations, the Kurchatov generating station at Beloyarsk, the Voronezh station, etc.). The following conclusions may be arrived at on the basis of the above:

1. At the present state of the art of reactor design, building of containment shells is expensive and in many cases superfluous overdesign;

2. Concrete biological shielding can handle the job of preventing release of radioactive elements in case of a reactor accident, without incurring additional cost penalties. In that case, taking account of the design internal overpressure, it is usually necessary to simply strengthen the reinforcement of the concrete in the lateral and top biological shielding and to make sure that the reactor chamber is leak-proof. This measure is taken in an experimental power reactor now under construction in the USSR, and will undoubtedly provide for sufficiently reliable localization of radioactivity release within the reactor chamber in case of an explosion [14].

3. Erection of special containment structures around the reactor may be justified only in the case where, in the first instance, the reactor is an experimental type and is prone to an explosive accident both with respect to basic design and with respect to the nature of the moderator or coolant employed; in the second instance, if the design of the biological shielding does not enable the latter to localize radioactive elements which might become ejected in the event of an accident to the area within the reactor hall; and in the third instance, if the reactor is situated so close to populated localities or buildings occupied by personnel that these could become targets of radioactive contamination in the event of an explosion.

4. In certain cases, the presence of the protective containment shell may even aggravate the danger of explosion. For example, the release of hydrogen contained in the pressurized coolant water by the explosion may result in the formation of an explosive mixture with the air trapped within the containment shell [10].

Taking into account the theoretical possibility of destruction or damage to the steel containment structure by missile fragments of equipment, parts, or tubes torn loose in a reactor explosion, recent practice has added the feature of a reinforced-concrete inner lining of 0.6 m thickness to provide an added safeguard inside the steel containment shells, in some designs. However, even this approach cannot fully guarantee that the steel shell and the reinforced-concrete lining will not be penetrated by fragments of equipment, since the calculated weights of such "missiles" and the predicted velocity imparted to them by the explosion are entirely arbitrary.

In the design of the USA package power reactor APPR-1, a power plant with a pressurized-water-cooled water-moderated reactor rated at 10 Mw thermal power, it was found through theoretical calculations that potential "missiles" could attain speeds up to 15 m/sec while traversing a path of 12 m inside the containment volume. Reinforced-concrete lining 0.60 m thick would under such conditions be pierced by such "missiles" as pieces of piping, valves 50 mm in diameter, pieces of control rods, etc. [10].

One positive quality that may be attributed to an interior reinforced-concrete lining is the fact that it provides some additional biological shielding, and another is that it limits the temperature rise of the steel shell in case of an accident, and consequently diminishes the attendant thermal stresses. However, the usual requirements

imposed on the biological shielding around the reactor and the steam generators for the safety of the operating personnel present inside the containment-shell volume during the operation of the reactor makes any additional shielding such as provided by the reinforced-concrete lining of the containment shell superfluous. The thermal stresses in the containment shell are not determining factors in deciding shell thickness, in most cases. Moreover, practice in building and placing this type of reinforced-concrete structure has revealed great difficulties in erecting and reinforcing them. The above-mentioned source [10] confirms the fact, based on experience in designing containment shells, that linings of reinforced concrete are costly in most cases, and that internal concrete barrier shields not integral with the steel containment structure are more practical. These concrete shields are constituted in some cases by the walls of the reactor biological shielding, by the heat exchangers, or by other radioactive equipment forming, together with the reactor top shield, pressure-tight chambers capable of containing, within certain limits, any radioactive release (in the event of a reactor explosion) and also capable of localizing the "missile" fragments which might be sent flying by such an explosion.

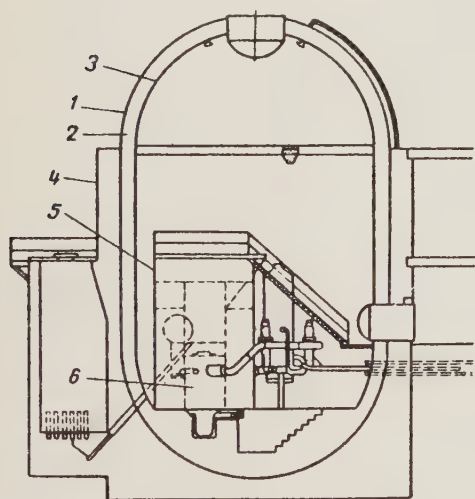


Fig. 8. Vapor containment shell of the APPR-1 reactor; 1) containment shell; 2) concrete lining; 3) steel shell; 4) secondary shield; 5) primary shield; 6) pressurized-water reactor.

The striving to nullify the probability of "missiles" being torn loose by a reactor explosion has decided the choice of material for the pressure vessel and piping of the primary loop in the case of the reactor for the US "Yankee" nuclear power station, where carbon steel lined with stainless steel is employed in the pressure vessel, and tough austenitic stainless steels are used for the ducting. Nevertheless, this reactor setup is buttressed by a "missile" shield incorporating concrete liners surrounding the reactor [9]. The APPR-1 package power reactor, with a steel containment shell backing up an external concrete biological shield 0.915 m thick, has in addition an interior reinforced-concrete "anti-missile" lining 0.60 m thick, which in essence forms part of the reactor biological shield. This lining is encased by a sheet steel envelope designed to facilitate deactivation of the reactor house (Fig. 8). Erection of the steel containment shell is no easy job, and will not be undertaken in future projects, as stated earlier. In conclusion, it is well to note that the variant of "missile" fragments flying loose in all directions in the event of a reactor explosion is highly improbable and the approach must be geared to eliminating the possibility of such missiles flying loose by judicious engineering of the basic equipment, pipe-work, fastening and hold-down devices. In giving thought all the same to the problem of protecting the steel shell against impact of these missiles, reliance should be placed on the concrete structure of special inner linings of reinforced concrete. Linings of this type may prove

feasible only in those rare cases when it is decided not to place a biological shield directly around the reactor and ancillary equipment, and to make use of only an outer shield which in this case will incorporate the anti-missile lining.

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MECHANICAL PROPERTIES AND MICROSTRUCTURE OF CERTAIN CONSTRUCTION MATERIALS AFTER NEUTRON IRRADIATION

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The article presents data on the influence of irradiation during normal operation of the reactor at the First Atomic Power Station on the mechanical properties and microstructure of a number of austenitic, ferrite, ferritemartensitic steels and molybdenum. The authors demonstrate a sharp falloff in the plasticity of the austenitic and ferrite steels and molybdenum as a result of irradiation with a total flux of $(0.9-3.4) \cdot 10^{20}$ neut/cm² at a temperature of 450-650° C. It is found that the causes of steel and molybdenum becoming brittle differ from one another in their nature. The information presented may prove useful in the planning of new atomic power plants.

The intent of the present paper was to find out the degree of influence of the operating conditions in the reactor at the First Atomic Power Station on the mechanical properties and microstructure of technically pure molybdenum and structural steels of the following brands: 1Kh18N9T, Kh20N14S2, 1Kh15N11M2C2T, 2Kh13, 1Kh13BMS2, and Kh10Yu5ST. Test steels, whose chemical compositions are listed in Table 1, were also investigated.

TABLE 1

Chemical composition of test steels

Steel brand	Content of elements, %							
	C	Cr	Nb	Mo	Si	Cu	W	V
1Kh13BM*	0,12	12,4	1,3	0,6	—	—	—	—
1Kh13B2M	0,10—0,15	12—14	1,5—2,0	0,8—1,2	—	—	—	—
1Kh13B2MSFB	0,10—0,15	12—14	1,5—2,0	0,5—1,0	0,5—1,0	—	0,2—0,5	0,2—0,5
2Kh17B2M	0,15—0,20	16—18	1,5—2,0	0,8—1,2	—	—	—	—
2Kh17B2MS2	0,15—0,20	16—18	1,5—2,0	0,8—1,2	1,5—2,0	—	—	—
2Kh17B2MS3	0,15—0,20	16—18	1,5—2,0	0,6—0,8	2,5—3,0	—	—	—
Kh17MS3D2*	0,03	18,0	—	1,0	3,0	1,75	—	—
2Kh17B2MS3P2	0,15—0,20	16—18	1,5—2,0	0,6—0,8	2,7—3,2	1,5—2,0	—	—

* Content by analysis.

To irradiate the samples we used the SUZ channel of the outer ring of the reactor at the First Atomic Power Station [1, 2]. In the empty graphite sockets of the channel, which were supposed to be in the active zone,

four cylindrical openings with a diameter of 18 mm were drilled out along the axis of the socket, and into these were placed the samples, contained in unsoldered quartz ampoules. The ampoules were in contact with the nitrogen atmosphere that filled the reactor pile.

The temperature was controlled by thermocouples located at three points of the active zone of the channel, one high, one middle position, and one low. The temperatures in between were determined according to the distribution of heat liberated inside the graphite pile over the height of the reactor. Depending on the location of the samples, as well as the operating power of the reactor, the temperature thus determined was found during the irradiation to lie within the interval 450-650° C. Heat liberation in the samples was not accounted for.

In preparation a part of the samples was stamped out of sheet material, the rest being cut from rods. The corresponding heat treatment was performed in vacuum ($\sim 1 \cdot 10^{-3}$ mm Hg). Cold plastic deformation was realized (after quenching of the steels) by drawing.

The number of test samples for each point was equal to from three to five. The testing was done at room temperature.

The total neutron fluxes shown further in Tables 2-5 are given for thermal neutrons and are approximate. It is assumed that the number of neutrons of higher energy amounts to $\sim 25\%$ of the number of thermal neutrons.

The mechanical properties of austenitic steels before and after irradiation are shown in Table 2. From the data obtained it follows that the initial state of the material plays a significant role in the change in properties during irradiation. For example, the microhardness of quenched steel 1Kh18N9T after irradiation increased by 65 kg/mm², the relative elongation of the quenched steels Kh20N14S2 and 1Kh15N11M2S2T decreased by a factor of three or more, and for steel 1Kh18N9T by approximately 20%.

TABLE 2

Mechanical properties of austenitic steels

Initial state	Steel brand	Total flux $\times 10^{-20}$ neutrons/cm ²	Irradiation temp, °C	Yield strength, kg/mm ²		Relative elongation,		Microhardness, kg/mm ²	
				Before irrad.	After irrad.	Before irrad.	After irrad.	Before irrad.	After irrad.
Quenching from 1100° C in water	1Kh18N9T	3.4	~650	57	58	39	31	138	203
	Kh20N14S2	2.4	~500	56	53	45	17	156	199
	1Kh15N11M2S2T	3.4	~650	56	54	45	11	156	187
	1Kh15N11M2S2T	2.7	~550	68	63	46	15	185	225
Quenching from 1100° C and cold deformation*	1Kh18N9T	3.4	~650	88	74	16	23	317	265
	Kh20N14S2	1.2	~500	96	91	16	17		
	1Kh15N11M2S2T	3.1	~600	139	110	4.4	8.8	432	412
Quenching from 1100° C, cold de- formation* and 3-hour anneal at 850° C	1Kh18N9T	3.4	~650	65	63	26	25	209	195
	Kh20N14S2	0.9	~450	69	67	33	20		
	1Kh15N11M2S2T	2.7	~550	86	82	20	11	273	283
The same with 10-hour anneal at 650° C	1Kh18N9T	2.4	~500	64	69	29	22	180	236
	Kh20N14S2	3.4	~650	64	64	29	31	180	178
	1Kh15N11M2S2T	1.0	~500	71	65	35	18	254	277
	1Kh15N11M2S2T	2.7	~550	87	86	15	13	286	300

*The shrinkage for steel 1Kh15N11M2S2T was $\sim 90\%$, for the rest $\sim 30\%$.

Figure 1 shows the microstructure of austenitic steel Kh20N14S2 in the quenched state before and after irradiation. The same changes in structure are noted for steels 1Kh18N9T and 1Kh15N11M2S2T. It is noteworthy that the quenched austenite grain assumed a more uniformly circular shape after irradiation at 500-650° C, while at the grain boundaries carbides were precipitated. The same change of structure is also observed after the action of a single temperature and, consequently, is not the result of the specific action of the neutron field. It is known

that being held in the region of very high temperatures gives rise to aging (dispersion hardening) of austenitic steels. With the simultaneous action of a neutron field and temperature, the aging process is probably accelerated [3].

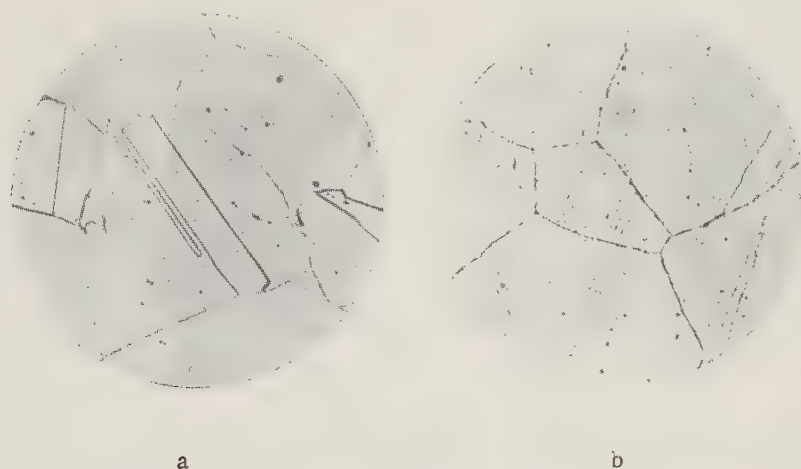


Fig. 1. Microstructure of steel Kh20Ni4S2 in the quenched state ($\times 325$): a) Before irradiation; b) after irradiation at a temperature of 650°C by a total flux of $3.4 \cdot 10^{20}$ neutrons/cm².

As a result of the work of various investigators [4] it is presently known that temperatures of the order of those considered in the present article are sufficiently high for the indicated steels from the viewpoint of being able to retain in these a large fraction of primary radiation dislocations. Nevertheless, since dislocations during the time of neutron bombardment must arise at any temperatures (although with an increase in the temperature at which bombardment is taking place the defects that appear will be annealed out in an ever-increasing number and at an ever-increasing rate) these dislocations should effect those processes which occur at these temperatures in the nonirradiated steels. With increasing temperature the relative participation of the neutron flux intensity should increase.

For cold-deformed steels a reduction in yield strength and hardness, as well as an increase in relative elongation are observed as the result of irradiation.

In the present instance there occur simultaneously in the steels two processes, which have opposite effects on their properties: relief of cold hardening by anneal and aging accompanied by the separation of hardening phases. There is no doubt that the observed dehardening of cold-deformed steels was the result of the former process.

It is known [5] that preliminary cold deformation augments the subsequent aging of steels. Figure 2 illustrates the austenitic decomposition of cold-deformed ($\sim 30\%$ deformation) steel Kh20Ni4S2 as the result of irradiation at a temperature of $\sim 500^{\circ}\text{C}$. From a comparison of Fig. 2 and Fig. 1b we can see the difference in the degree of austenitic decomposition and in the distribution of separations between the quenched and cold-deformed steels.

Stabilization of the structure of cold-deformed austenitic steels by anneal at 850°C for a period of 3 hours or two-stage anneal at 850°C for 3 hours and at 650°C for 10 hours leads to a reduction in the degree of decline in plasticity as the result of irradiation. The yield strength of the stabilized steels as the result of irradiation was not altered significantly.

Consequently, it may be concluded that the observed change in the properties of austenitic steels as the result of irradiation by a total flux of $0.9\text{--}3.4 \cdot 10^{20}$ neutrons/cm² at temperatures of $500\text{--}650^{\circ}\text{C}$ are connected primarily with the nonequilibrium state of the material and with the processes taking place in it.

A neutron field does not exert any essential influence on the mechanical properties of ferrite and ferrite-martensitic steels (Table 3). This is clearly evident from a comparison of the properties of the steels tested after irradiation at 600°C and after being held at 600 and 700°C without irradiation* (Table 4).

* The information relating to the properties of the test steels after prolonged exposure to temperatures of 600 and 700°C without irradiation was obtained by M. D. Abramovich.

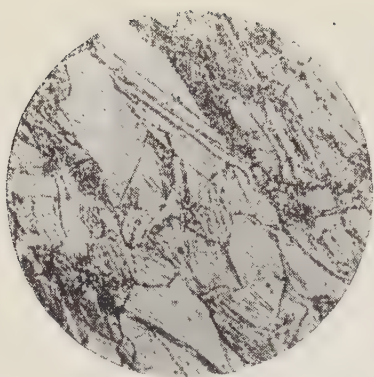


Fig. 2. Microstructure of quenched cold-deformed steel Kh20N14S2 after irradiation at a temperature of $\sim 500^\circ\text{C}$ by a total flux of $1.2 \cdot 10^{20}$ neutrons/cm².

Unlike the austenitic and ferrite steels, in the region of the temperatures investigated there was no appreciable aging accompanied by separation. But in ferrite 17%-chrome steels containing a large quantity of silicon there occurred as the result of irradiation a very strong drop, unobserved in ferrite steel containing no silicon, in the magnitude of the relative elongation. This change in plasticity was evidently caused by those factors (e. g., the x-ray detected separation of the σ -phase) which give rise to brittleness of the given steels after long exposure to temperatures of about 600°C without irradiation.

In 13%-chrome steels belonging to the ferrite-martensitic class, heat-treated prior to irradiation to the stable ferrite-perlite structure, no significant changes in the properties were observed after irradiation.

TABLE 3

Mechanical properties of ferrite-martensitic steels

Steel brand	Initial state	Total flux 10^{20} neutrons/cm ²	Irradiation temp, $^\circ\text{C}$	Yield strength, kg/mm ²		Relative elongation, %		Microhardness, kg/mm ²	
				Before irrad.	After irrad.	Before irrad.	After irrad.	Before irrad.	After irrad.
2Kh13	1-hour anneal at 750°C ...	0.9	~ 450	59	60	28	24	234	230
1Kh13BMS2	1-hour anneal at 750°C ...	0.9	~ 450	65	67	25	23	225	254
Kh10Yu5ST (ferrite)	As delivered	2.7	~ 550	68	71	15	13	—	—

TABLE 4

Mechanical properties of test steels

Steel brand	Yield strength, kg/mm ²				Relative elongation, %			
	Initial state*	After holding at 600°C for 1560 hr	After holding at 700°C for 1000 hr	After irra- diation at $\sim 600^\circ\text{C}$ (total flux $3.1 \cdot 10^{20}$ neutrons/ cm ²)	Initial state*	After holding at 600°C for 1560 hr	After hold- ing at 700°C for 1000 hr	After ir- radiation at $\sim 600^\circ\text{C}$ (total flux $3.1 \cdot 10^{20}$ neutrons/ cm ²)
1Kh13BM.	60	58	54	51	23	23	20	20
1Kh13B2M.	59	56	53	48	21	22	24	23
1Kh13B2MSFV.	63	61	52	55	18	20	22	19
2Kh17B2M.	60	56	55	52	20	21	23	19
2Kh17B2MS2.	73	72	68	72	21	12	23	8.0
2Kh17B2MS3.	83	75**	69	72	19	2.0**	3.0	2.5
Kh17MS3D2.	76	73	66	86	18	3.0	9.0	4.3
2Kh17B2MS3D2.	81	74	67	84	18	1.5	5.0	1.2

*The 13 % steels were annealed at 750°C for 2 hr, the rest were normalized from 900°C .

**Held for 1000 hr.

TABLE 5

Mechanical properties of molybdenum before and after irradiation

Molybde- num batch	Initial state	Total flux $\times 10^{-20}$ neutrons/ cm^2	Irradia- tion tem- perature, $^{\circ}\text{C}$	Yield strength, kg/mm^2		Relative elong- ation, %		Microhardness, kg/mm^2	
				Before irrad.	After irrad.	Before irrad.	After irrad.	Before irrad.	After irrad.
I	30-min anneal at 1250°C	1.2	~ 550	65	84	21	2.6	—	—
II	2-hour anneal at 1000°C	2.0	~ 450	72	110	7	1.7	187	366

Table 5 shows the change in properties of technically pure molybdenum after irradiation. The increase in strength and hardness of molybdenum as the result of irradiation and the sharp reduction in relative elongation strongly depend on the initial state of the molybdenum, as noted earlier [6].

It appears that in connection with the high melting temperature and high elastic properties of molybdenum, no amount of exposure to temperatures of $450\text{--}550^{\circ}\text{C}$ without irradiation will give rise to any processes that are able to strongly modify the properties of molybdenum. The conclusion may therefore be made that radiation dislocations arising during irradiation and influencing the mechanical properties are retained even at temperatures of $450\text{--}550^{\circ}\text{C}$. In steels subjected to the same conditions a large part of the radiation dislocations apparently succeed in being annealed away.

The authors wish in conclusion to express their gratitude to E. V. Chermashentsev and A. Ya. Ladygin for their part in the work undertaken.

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EXTRACTION OF URANIUM FROM SOLUTIONS AND PULPS

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Extraction processes are finding increasing application for the processing of uranium from raw materials. The high selectivity and the fluid aggregate state of the extraction agents give this method advantages over the previously known processes.

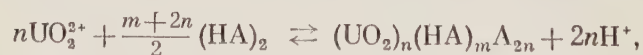
This article gives data on the extraction of uranium from sulfate, nitrate, chloride and phosphate solutions and pulps, most frequently encountered in the hydrometallurgy of uranium. The extraction agents suitable for industrial use are the esters of carboxylic acids, the esters of phosphoric and phosphinic acids and also liquid cation-exchange materials and anion-exchange materials, in a number of cases (in the extraction of uranium from colored solutions) have advantages over solid ion exchange materials. A system is described for the extraction of uranium from dense ore pastes.

Introduction

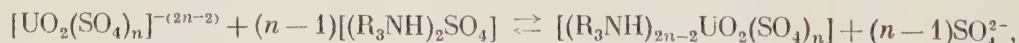
Extraction methods were used for uranium at the start of the development of the uranium industry. In hydrometallurgy, extraction was used at the stage of obtaining pure uranium compounds. At a later stage, together with the development of sorption processes for extracting uranium from solutions and ore pulps, work was carried out on extraction from nitrate, sulfate, chloride desorption and other solutions and ore pulps of varying thickness.

By drawing an analogy between the mechanism of absorption of sorbents and extraction agents it is possible to find and use new classes of organic solvents which are also liquid ion-exchange materials.

In a number of cases the difference in the processes of sorption and extraction consists mainly in the difference between the aggregate state of the organic phase (absorber). For example, the mechanism of extraction by alkyldithiophosphoric, alkylphosphoric and other acids is similar to the mechanism of sorption by cation exchange materials:



and the mechanism of extraction by alkylamines is similar to the mechanism of anion exchange:



where $\underline{n}$ is equal to 2 or 3.

The difference in the aggregate state of extraction agents and sorbents provides new possibilities in the use of liquid absorbents. Compared with solid absorbents, liquid absorbents have a much higher rate of absorption and capacity, and hence there is a greater degree of concentration of uranium during extraction from dilute solutions. In particular, liquid absorbents can be used for extraction from anhydrous ore materials whereas sorption processes can only be used with ore pulps containing not more than 40% solid materials. In a number of cases solid extraction agents are of interest, for example, porous active sorbents impregnated with extraction agents (gels and activated carbons).

TABLE 1

The Solubility of Esters of Acetic Acid in Water and Selectivity with Regard to Uranium

Extraction Agent	Chemical Formula	Molecular weight	Solubility in water at 20°C, %	Total of impurities in U_3O_8 , %
Isopropyl acetate	$CH_3COOCH(CH_3)_2$	102	3	2.6
Butyl acetate	$CH_3COOC_4H_9$	116	0.6	0.15
Isoamyl acetate	$CH_3COO(CH_2)_2CH(CH_3)_2$	172	0.25	0.06

TABLE 2

Selectivity of Certain Extraction Agents for Uranium

Extraction Agent	Chemical Formula	Total of impurities in U_3O_8 , %
Isooctyl alcohol	$CH_3(CH_2)_3CH(CH_2H_5)CH_2OH$	1.7
Dibutyl ether	$(C_4H_9)_2O$	0.2
Isoamyl acetate	$CH_3COO(CH_2)_2CH(CH_3)_2$	0.06
Isooctyl acetate	$CH_3COOCH_2CH(C_2H_5)(CH_2)_3CH_3$	0.05
Tributyl phosphate	$(C_4H_9O)_3PO$	0.5

Extraction of Uranium from Solutions

Studies have been made of the extraction of uranium by different classes of organic compounds (alcohols, ethers, ketones, diketones and their halogen derivatives, esters, carboxyl, phosphoric, phosphinic, alkyldithiophosphoric acids, aliphatic amines, etc). This section will only deal with the properties of a few of the solvents.

The extraction of uranium by alcohol, ethers and esters of carboxylic acids. The extraction agents of this group extract uranium from nitrate solutions and can only be used in the presence of salting-out materials since under ordinary conditions they have a low distribution coefficient ($K_p = 0.2-0.7$). With increase in the length of the aliphatic radical, K_p of the solvents decreases in a definite order for each homologous series.

It is characteristic that with increase in the length of the hydrocarbon chain by a CH_2 group in the acid radical of the carboxylic esters, K_p is reduced more noticeably than with a corresponding increase in the length of the hydrocarbon chain in the alcohol group. Increase in the temperature of the solutions leads to a decrease in K_p .

The value of carboxylic esters and ethers consists of their high selectivity with respect to uranium. It has been found that with increase in the molecular weight of the solvent the selectivity of uranium extraction increases.

It can be seen from Table 1 that the selective capacity of acetic esters increases with their solubility in water. The selectivity of extraction agents is reduced in the order: esters of carboxylic acids, ethers, alcohols. Their data are given in Table 2; for comparison purposes information is given on tributyl phosphate (TBP) which has been extensively used in the extraction of uranium.

K_p for the extraction of uranium by alcohol, ethers and esters of carboxylic acids depends strongly on the concentration of salting-out materials in the aqueous solution. This relationship is shown in Fig. 1 for isoamyl acetate and dimethyl phthalate. The extraction of uranium by these types of solvents is strongly inhibited by sulfate, phosphate, arsenate and other ions.

In a number of cases, including extraction of uranium from thick nitrate pulps, effective use can be made both of carboxylic esters and ethers.

The literature [1] indicates the possibility of effective use of such compounds as tetrahydrosilvan and tetrahydropyran for the extraction of uranium from nitrate solutions with small amounts of salting-out materials.

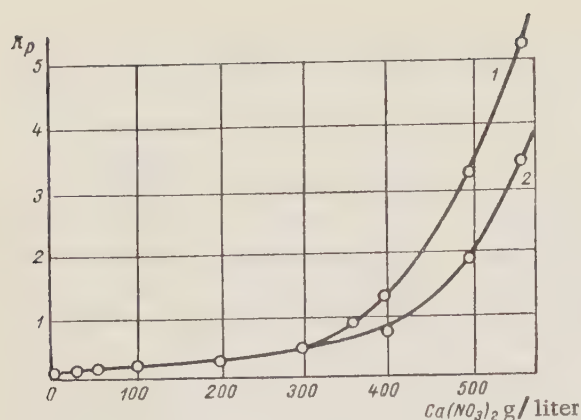


Fig. 1. The salting-out action of $Ca(NO_3)_2$ in the extraction of uranium by isoamyl acetate (curve 1) and dimethyl phthalate (curve 2).

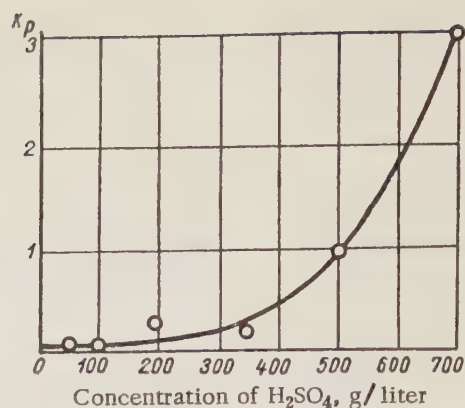


Fig. 2. Extraction of uranium by 100% TBP from sulfuric acid solution, uranium concentration 10 g/liter; ratio of volume of organic phase to water, 10.4.

Esters of phosphoric acid. Trialkyl phosphates are a well-known class of selective solvents for uranium. K_p of the trialkyl phosphates increases with increase in the hydrocarbon radical up to $C_5 - C_6$, then gradually decreases. The nature of the hydrocarbon radicals is then of decisive importance. For example, the triaryl phosphates are almost unable to extract uranium.

Increasing the temperature for ethers and esters of carboxylic acids leads to a reduction in K_p . The selectivity of trialkyl phosphates increases with the molecular weight of the extraction agent (Table 3). The introduction of salting-out materials (nitric acid and nitrate salts) causes an increase in K_p of uranium. The extraction is strongly inhibited by sulfates, arsenates, phosphates and fluorides. It is of interest that TBP also extracts uranyl sulfate but only from solutions with a high concentration of sulfuric acid. As can be seen from Fig. 2, with a sulfuric acid concentration of 700 g/liter, K_p is equal to three for extraction by 100% TBP.

TABLE 3

Extraction Agent	Chemical Formula	Molecular weight	K_p (for 1 M solution of extraction agent in kerosene)	Total of impurities of U_3O_8 , %
Tripropyl phosphate	$(C_3H_7O)_3PO$	224	7.4	—
Tributyl phosphate	$(C_4H_9O)_3PO$	266	8.3	0.5
Triamyl phosphate	$(C_5H_{11}O)_3PO$	310	9.3	0.3
Trihexyl phosphate	$(C_6H_{13}O)_3PO$	352	9.6	—
Trioctyl phosphate	$(C_8H_{17}O)_3PO$	434	9.3	0.13
Triphenyl phosphate	$(C_6H_5O)_3PO$	326	0.025	—
Tricresyl phosphate	$(CH_3C_6H_4O)_3PO$	368	0.0015	—

Note: K_p was determined at equilibrium conditions in equal volumes of the phases. The initial solution contained 30 g/liter of uranium and 25 g/liter of nitric acid.

TBP was used in the extraction and purification of chemical concentrates and also in connection with the extensive development of sorption technology in the purification of nitrate desorption (regeneration) solutions of uranium (Fig. 3).

Trialkyl phosphates can be used as plasticizers in films of cellulose esters, polymonochlorostyrene and other materials. It was found that these solid extraction agents can extract uranium from nitrate solutions. For example, 1 g of a cellulose film containing 16% TBP absorbs 35 mg of uranium from solution providing the initial solution contains 5 g/liter uranium, 200 g/liter nitric acid and 300 g/liter ammonium nitrate.

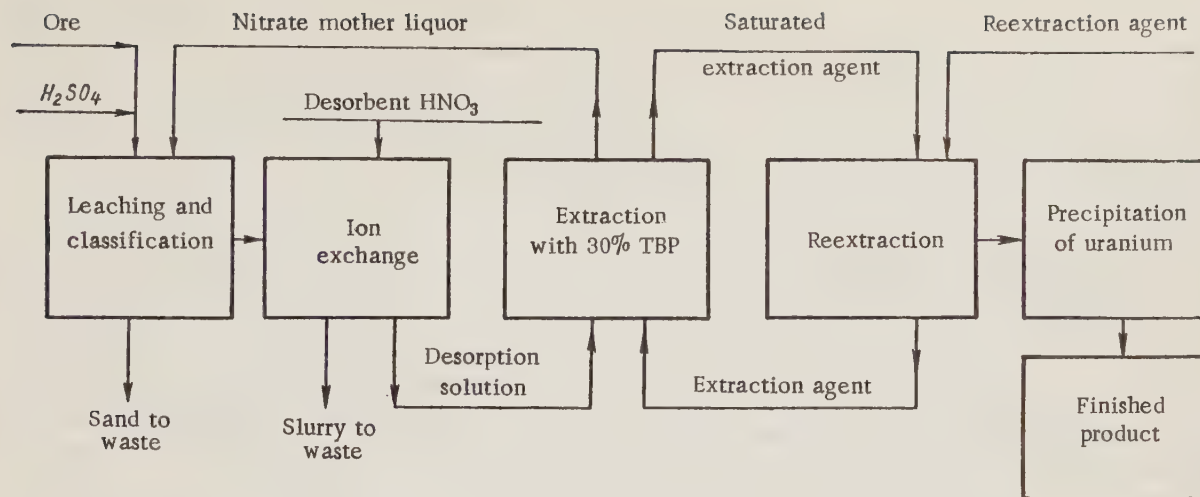


Fig. 3. Flow sheet for the extraction purification of nitrate desorbed solutions of uranium.

Diisoamyl ester of methylphosphinic acid (DAMPA). DAMPA [$i(C_5H_{11}O)_2POCH_3$] is one of the lower members of the alkylphosphonate series. The representatives of this series with large molecular weights, for example the dihexyl ester of hexylphosphinic acid or the dioctyl ester of methylphosphinic acid are also very interesting since they are less soluble in water and very effective in the extraction of uranium.

DAMPA is a transparent colorless liquid of specific gravity 0.953, refractive index 1.4284. The solubility of DAMPA in water is 0.3 g/liter, boiling point 256°C. DAMPA extracts uranyl nitrate and nitric acid, which is described in [2]. DAMPA is readily soluble in various organic solvents and can be used to extract uranium from nitrate and chloride solutions. In the extraction of uranium from a 1 M solution of nitric acid, K_p for 100% DAMPA reaches 100 with an initial concentration of uranium in the solution of 10 g/liter. Increase in the nitric acid concentration leads to a considerable increase in K_p . It has been shown that DAMPA has fairly good selectivity. In the extraction of uranium from nitrate solutions, iron is extracted to a very small extent and aluminum and calcium are extracted in small quantities.

In the extraction from chloride solutions K_p increases considerably with increase in the hydrochloric acid concentration (Fig. 4).

Sulfate and phosphate ions inhibit the extraction of uranium with DAMPA. Figure 5 shows a curve characterizing the inhibiting action of phosphoric acid.

The presence of small amounts of nitric acid (15-20 g/liter) considerably increases the extraction of uranium from phosphate solutions. For example, in the extraction of uranium from a solution containing 300 g/liter of phosphoric acid, $K_p = 0.23$. When uranium is extracted by a 10% solution of DAMPA in kerosene, equilibrium is established after 2-3 minutes. The reextraction of uranium from DAMPA can be carried out by 5-10% solutions of sulfuric acid and ammonium carbonate and also by other reagents.

Triooctyl phosphinoxide (TOPO). TOPO [$(C_8H_{17})_3PO$] forms stable complex compounds with uranium. The mechanism of extraction has not been fully studied although there are reasons to assume that compounds such as TOPO form complexes with uranium similar to TBP. TOPO is readily soluble in carbon tetrachloride, aromatic hydrocarbons, alcohols, esters and somewhat less soluble in petroleum ether and kerosene. A 0.1 M solution of TOPO in kerosene can be prepared. With increase in temperature the solubility of TOPO in all solvents increases noticeably. The solubility of TOPO and com-

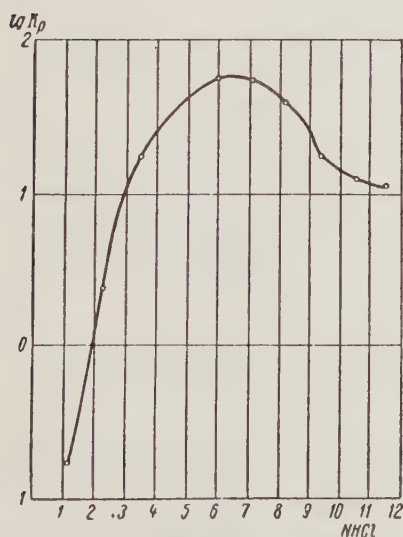


Fig. 4. Extraction of uranium by a 10% solution of DAMPA in kerosene from chloride solutions (initial solution 3.5 g/liter uranium, ratio of volumes of organic phase to aqueous phase equal to 1).

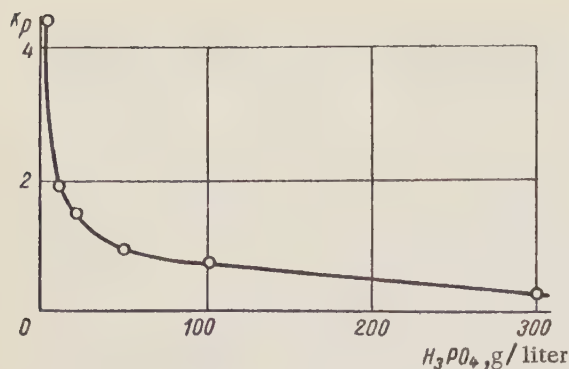


Fig. 5. The effect of phosphoric acid on the extraction of uranium by a 10% solution of DAMPA in kerosene from solutions containing 20 g/liter of nitric acid.

that at low concentrations of uranium K_p maintains a large value. This property means that TOPO can be used for the extraction of uranium from very dilute solutions.

Carbonate solutions can be used in the reextraction of uranium from TOPO. For example, a 10% solution of ammonium carbonate heated to 40°C in one operation lasting 1-2 minutes reextracts up to 99.5% uranium with a ratio of volumes of organic phase to aqueous phase equal to 4. Reextraction of uranium by 10 N hydrochloric acid is less effective. TOPO has good chemical stability. Its extraction properties hardly change even after repeated use. The prolonged action of concentrated acids does not produce any visible changes or any changes in the extraction properties of TOPO.

The addition of alkyl phosphates, alkyl phosphonates, alkyl phosphinates and particularly alkyl phosphinoxides to solutions of alkyl phosphoric acids considerably increase the value of K_p in the extraction of uranium.

Dialkyl phosphites. Studies have been made of various dialkyl phosphites $[(RO)_2P(OH)]$ with hydrocarbon radicals from C_4 to C_8 . The methods for preparing dialkylphosphites were based on the reaction of phosphorus trichloride with the appropriate alcohol.

The studied dialkyl phosphites were soluble in hydrocarbons, their chlorine derivatives, ethers, alcohols and ketones. Compounds of uranium with the lower dialkyl phosphites (with C_4 - C_5) were insoluble in aliphatic hydrocarbons. The solvate of uranium with dioctyl phosphite is readily soluble in kerosene.

Diisoalkyl phosphites containing the hydrocarbon radical C_4 - C_5 extract uranium from 0.3 N solutions of nitric acid with $K_p = 8$ -12 of uranium in the extraction of dialkyl phosphites increasing with increase in the concentration of nitric acid in the aqueous solution. K_p of dialkyl phosphites increases with increase in the molecular weight. For example, in the extraction of uranium by 1 M solutions of dialkyl phosphite from nitrate solutions containing 0.3 N HNO_3 , K_p for dibutyl phosphite is equal to 2, for diamyl phosphite 2.8 and for dioctyl phosphite -800.

In contrast to dibutyl and diamyl phosphites, dioctyl phosphite extracts uranium from sulfate solutions with high values of K_p (> 1000); with increase in the concentration of sulfuric acid from 50-100 g/liter there is a considerable reduction in K_p .

Alkyl phosphoric acids. Among the widely used extraction agents there are dialkyl phosphoric $(RO)_2P(O)OH$ and dialkyldithiophosphoric $(RO)_2P(S)SH$ acids, in particular di-(2-ethylhexyl) phosphoric and di-(2-ethylhexyl) dithiophosphoric acids. Dialkylphosphoric compounds were described in [3, 4] and have been used in certain plants in the US. Alkyldithiophosphoric acids are of interest since they are cheap and readily available extraction agents for uranium and can be used when treating chloride, sulfate and phosphate solutions. Compounds of this type are liquid cation-exchange materials. Molecules of dialkyldithiophosphoric acids are apparently dimerized like the molecules of dialkylphosphoric acids and react with the uranyl ions as an exchange reaction described at the beginning of the article.

pounds of uranium with TOPO are similar. The impurities which are normally present in industrial solutions, except trivalent iron and chromium, are extracted very little by TOPO. TOPO extracts uranium from nitrate and chloride solutions. Phosphoric acid reduces the extraction of uranium by TOPO much more strongly than other impurities.

The introduction of complex-forming reagents such as citrate and oxalate ions considerably increases the extraction of uranium from solution. In the extraction of uranium by a 0.1 M solution of TOPO in kerosene from nitrate solutions, $K_p = 900$. Increasing the concentration of nitric acid in nitrate solutions causes a very small increase in K_p . When the concentration of nitric acid is increased to 200 g/liter there is a certain reduction in K_p .

In the extraction of uranium by TOPO, equilibrium is attained after 0.5-1 min. It has been shown

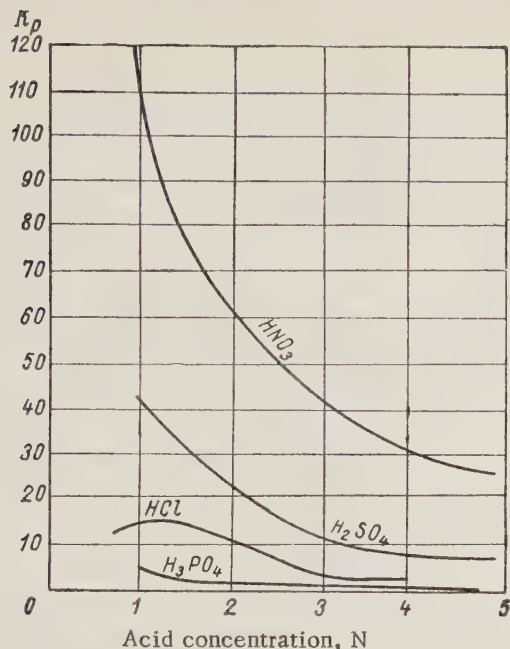


Fig. 6. Relationship between K_D of uranium and the concentration of mineral acids in the extraction by a 10% solution of dioctyldithiophosphoric acid in kerosene (initial solution 1 g/liter of uranium, ratio of volumes of organic phases to aqueous phases equal to 0.2).

Figure 6 gives curves for the dependence of K_D of uranium in the extraction by a 10% solution of dioctyldithiophosphoric acids in kerosene on the concentration of various mineral acids in the aqueous solution. For the extraction of uranium from phosphate solutions with a high concentration of phosphoric acids which are obtained in the acid decomposition of uranium-containing phosphorites use is made of the mono- and dialkylpyrophosphoric acids ($H_2RP_2O_7$, $HR_2P_2O_7$), which are efficient in extracting uranium. In particular, use is made of diisooctylpyrophosphoric and diisodecylpyrophosphoric acids with varying structures of the hydrocarbon radicals.

Amines. Long chain alkylamines and alkylarylamines extract uranium from sulfate solutions and concentrated solutions of hydrochloric and nitric acid in a similar manner to anion-exchange resins [5]. All the rules established for the sorption of uranium by anion-exchange materials from sulfate, nitrate, chloride, phosphate and other solutions can be used in the first approximation when considering processes of uranium extraction. In the extraction, use was made of primary, secondary and tertiary aliphatic and aliphatic-aromatic amines which were insoluble in water. The solvent for the amine compounds was a mixture of kerosene with alcohol. In view of the tendency of amines to form emulsions with aqueous solutions the content of solid particles in them should be at a minimum. To reduce the emulsification of amines with water it is best to use amines with branched aliphatic radicals. The emulsification can also be reduced by increasing the acidity and temperature of the solutions.

Long chain aliphatic amines have high chemical stability in acid and alkaline solutions. The selectivity of alkylamines with regard to uranium increases on transition from the primary to tertiary amino compounds. For example, the secondary long chain amines are somewhat more selective than the primary compounds but in contrast to the latter they extract trivalent iron in large quantities. When using secondary aliphatic amines it is therefore essential to reduce the trivalent iron. K_D of uranium depends to a considerable extent on the length and structure of the hydrocarbon radical. The solvents of alkylamines can also be benzene, carbon tetrachloride, chloroform, etc.

Table 4 gives K_D of uranium for various amines. A comparative evaluation of the extraction properties of alkylamines was carried out for decimolar solutions of amines in kerosene in the extraction of uranyl sulphate from pure solutions at pH = 1 and from solutions containing 50 g/liter of sodium sulfate for pH = 1 and a uranium concentration in the initial solution of 1 g/liter.

The following conclusions can be drawn from data on the dependence of K_D on the molecular weight and structure of hydrocarbon radicals for a number of amine extraction agents (Table 4):

- 1) for the homologous series of n-alkylamines, K_D of uranium increases with the molecular weight of the amines and has a maximum value for trioctylamine;
- 2) K_D of asymmetric amines such as methylstearylamine and stearyldimethylamine, is comparatively small (5-10) and depends very strongly on the nature of the diluent;
- 3) when using alkylamines the degree of branching of the aliphatic radicals is very important. The deciding role is probably the spatial hindrances. For example, K_D of uranium for tri-n-octylamine is equal to 200, and for tri-(2-ethylhexyl)amine it is less than 1, which can be explained by the screening of the nitrogen atom by the ethyl radicals;

4) in extraction from sulfate solutions special attention should be paid to benzylheptadecylamine, for which K_p of uranium is equal to or greater than 1000;

5) increasing the concentration of alcohol in the solvent leads to a reduction in K_p .

In the extraction of uranium from sulfate solutions, nitrate, chloride and phosphate ions reduce the extraction properties of the amine extraction agents. With regard to their depressing action they can be placed in the following order: nitrate, chloride, phosphate and sulfate ions. They almost entirely inhibit the extraction properties of alkylamines with a concentration of nitrate and chloride ions of ~ 15 -30 g/liter. Reduction in the extraction properties of alkylamines by sulphate ions is observed to a noticeable extent only at very high concentrations of sulphates (~ 200 -300 g/liter) and at $\text{pH} \approx 4$.

Figure 7 shows the dependence of uranium extraction by a 0.1 M solution of tridecylamine in kerosene on the concentration of mineral acids in the initial solutions. As can be seen from this diagram, trialkylamines can be used to extract uranium from 6-10 M solutions of nitric and hydrochloric acids. Under these conditions, using trialkylamines, efficient separations can be made of a mixture of thorium and uranium, plutonium and uranium, and also a mixture of trans-uranium elements.

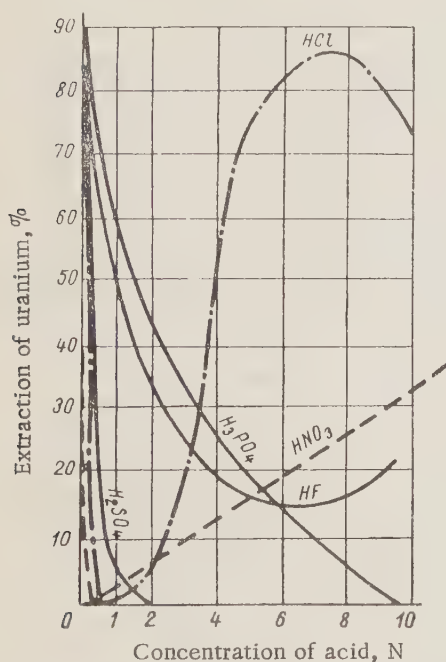


Fig. 7. Curves showing the dependence of extraction of uranium by a 0.1 M solution of tridecylamine in kerosene on the concentration of mineral acids in the initial solutions.

The maximum selectivity is shown by tertiary alkylamines, since they hardly extract bi- and trivalent iron, aluminum, copper, calcium and other impurities found with uranium. From sulfate solutions with $\text{pH} = 1$ -1.5 molybdenum is extracted together with uranium; however they can readily be separated during reextraction, when the uranium is reextracted with a 1 M solution of common salt at $\text{pH} = 1$ and molybdenum by a 5% solution of soda. Figure 8 shows the flow sheet for the extraction of uranium and molybdenum by tri-n-octylamine from a sulfate percolation solution.

For the reextraction of uranium from alkylamines the most efficient are solutions of chlorides, nitrates, soda and ammonium carbonate. The reextraction is entirely finished in 3-5 stages with a ratio of volumes of organic phase to aqueous phase equal to 6-8. During extraction from sulfate solutions the reextraction of uranium is best carried out with a 2.5-3 M solution of ammonium sulfate at $\text{pH} = 4.5$.

Extraction from solutions can therefore be accomplished by a large number of solutions with varying and often complex salt compositions. Due to their high selectivity, extraction processes have been used in the first place to purify uranium and also in those cases where the use of sorption processes causes difficulties, for example, when extracting uranium from concentrated solutions of phosphoric acid. When treating a number of solutions with low uranium content it is economically advisable to use sorption, based on the fact that losses of extraction agents are proportional to the volume of the solutions being processed. Methods have now been developed for the extraction of uranium from practically all types of solutions found in the uranium industry.

Extraction of Uranium from Thick Ore Pulps

The extraction of uranium from solutions is applicable mainly in those cases where colored solutions are obtained comparatively simply: in percolation, acid filtration or countercurrent decantation.

The extraction of uranium directly from pulps has a number of important advantages: it eliminates the need for classification of sands, the need to thicken slurries, repulping and filtration [7]. The most promising method for processing uranium ores is the extraction of uranium from thick ore pulps or pastes. The losses of the extraction agent depend to a large extent on the concentration of the solid material in the pulp (Fig. 9).

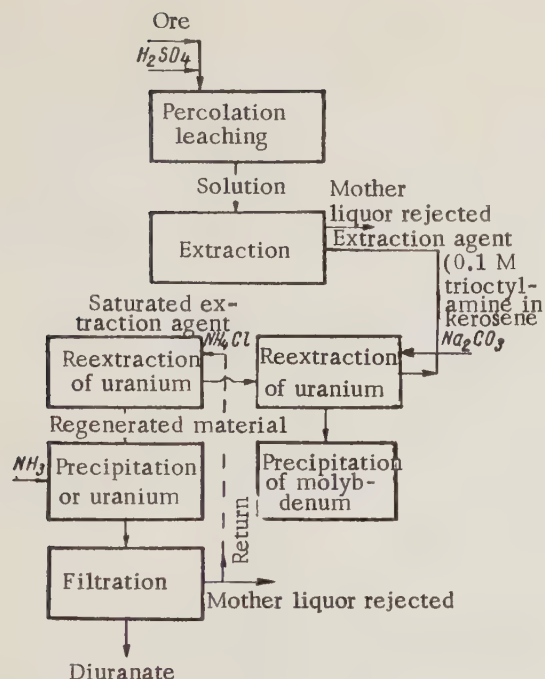


Fig. 8. Flow sheet for the extraction of uranium from percolation solutions by a 0.1 M solution of tri-n-octylamine.

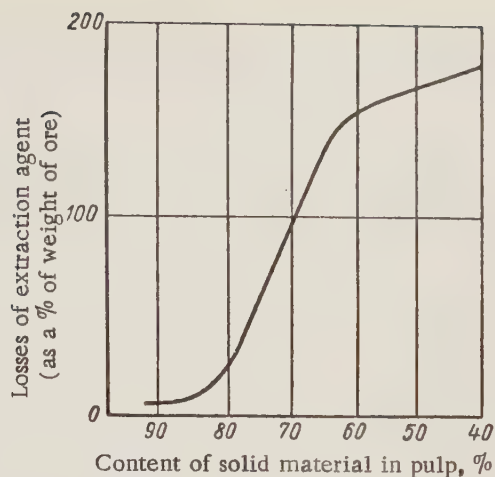


Fig. 9. Losses of extraction agent in the extraction of uranium from thick pulps.

It can be seen from Fig. 9 that the minimum losses of extraction agent are found in very thick ore pulps or pastes. The flow sheet for the extraction of uranium from ore pastes is given in Fig. 10. The non-aqueous extraction leaching of uranium is very promising for certain types of ores.

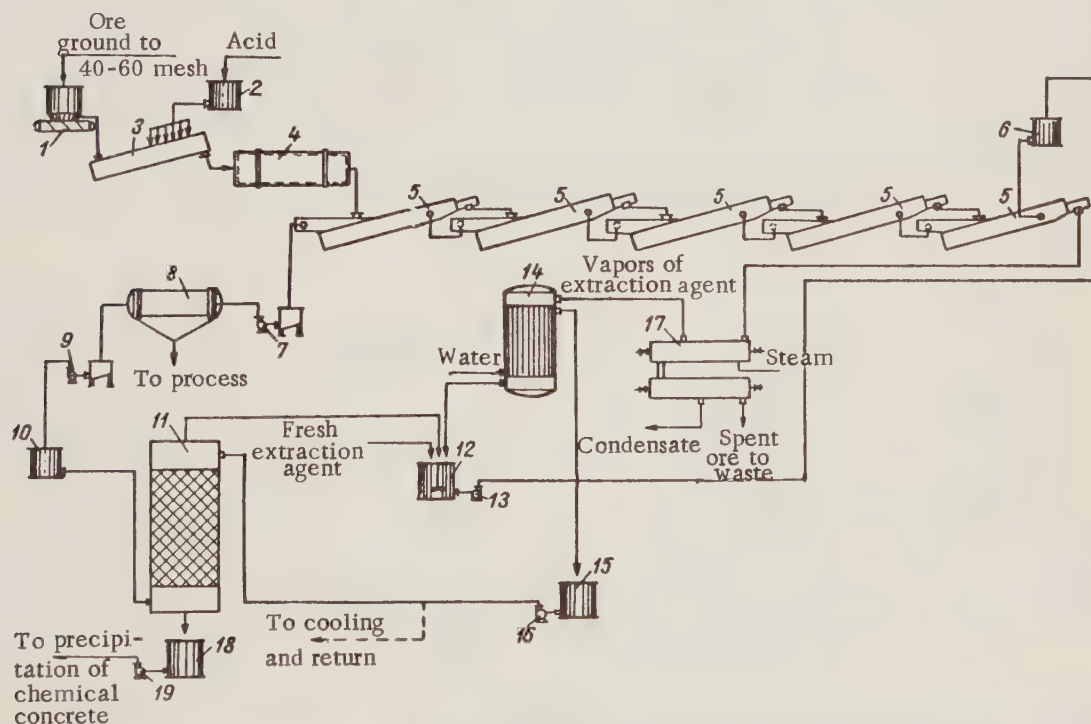


Fig. 10. Flow sheet for the extraction of uranium from ore pastes: 1) feeder; 2) tank; 3) apparatus for mixing; 4) apparatus for drying and cooling; 5) extraction fan; 6), 10) header tanks; 7), 9), 13), 16), 19) pumps; 8) filter press; 11) reextraction column; 12) contact vessel; 14) cooler; 15) tank; 17) apparatus for distillation of extraction agent; 18) collector.

TABLE 4

Extraction of Uranium by 0.1 M Solutions of Long Chain Amines

Amines	Chemical formula	Molecular weight	Solvent *	K _p of hexavalent uranium	
				1 g/liter of uranium (pH= 1)	1g/liter uranium and 50 g/liter SO ₄ ions (pH = 1)
Primary Amines					
n-Decylamine	C ₁₀ H ₂₁ NH ₂	157	50% alcohol , 50% kerosene	61.6	—
n-Dodecylamine	C ₁₂ H ₂₅ NH ₂	185		22.8	—
Secondary Amines					
Di-n-hexylamine	(C ₆ H ₁₃) ₂ NH	185	5 % alcohol , 95% kerosene	2.55	8.6
Di-n-octylamine	(C ₈ H ₁₇) ₂ NH	241		172.0	22.7
Di-(2-ethylhexyl)amine	(C ₈ H ₁₇) ₂ NH	241		110.0	—
Di-n-decylamine	(C ₁₀ H ₂₁) ₂ NH	297		64.0	21.7
Methylstearylamine	C ₁₈ H ₃₇ NHCH ₃	283	50% alcohol , 50% kerosene	10.0	—
N-benzylheptylamine	C ₆ H ₅ CH ₂ NHC ₇ H ₁₅	205	90% kerosene , 10% alcohol	12.1	63.0
N-benzyl-octylamine	C ₆ H ₅ CH ₂ NHC ₈ H ₁₇	219		154.0	154.0
N-benzylhexyl-(2-ethyl-amine)	C ₆ H ₅ CH ₂ NHC ₈ H ₁₇	219	90% kerosene , 10% alcohol	36.4	36.4
N-benzyl-n-nonylamine	C ₆ H ₅ CH ₂ NHC ₉ H ₁₉	233		44.7	—
N-benzyl-n-decylamine	C ₆ H ₅ CH ₂ NHC ₁₀ H ₂₁	247		41.0	—
N-benzyl-n-dodecyl-amine	C ₆ H ₅ CH ₂ NHC ₁₂ H ₂₅	275		23.0	11.5
N-benzyl-1-(3-ethyl-pentyl)-4-ethyloctyl-amine**)	C ₆ H ₅ CH ₂ NHC ₁₇ H ₃₅	345	kerosene	8000	—
Tertiary Amine					
Triisoamylamine	i(C ₅ H ₁₁) ₃ N	227	5% alcohol , 95% kerosene	0.048	0.147
Tri-n-hexylamine	(C ₆ H ₁₃) ₃ N	269		114.0	88.0
Tri-n-octylamine	(C ₈ H ₁₇) ₃ N	353		200.0	180.0
Tri-(2-ethylhexyl)amine	(C ₈ H ₁₇) ₃ N	353		1.0	1.0
Tri-n-decylamine	(C ₁₀ H ₂₁) ₃ N	437	100% alcohol	172.0	96.0
Octyldimethylamine	C ₈ H ₁₇ N(CH ₃) ₂	167		0.05	0.07
Decyldimethylamine	C ₁₀ H ₂₁ N(CH ₃) ₂	195		0.38	1.12
Stearyldimethylamine	C ₁₈ H ₃₇ N(CH ₃) ₂	297		5.10	3.55
N-benzyl-n-dioctyl-amine	C ₆ H ₅ CH ₂ N(C ₈ H ₁₇) ₂	331	5% alcohol , 95% kerosene	23.0	26.4
N-benzyl-n-dodecyl-amine	C ₆ H ₅ CH ₂ N(C ₁₀ H ₂₁) ₂	387		23.0	8.6
* The solvent was sulphonated kerosene and isooctyl alcohol.					
* *From the data of [6].					

* The solvent was sulphonated kerosene and isooctyl alcohol.

** From the data of [6].

In this method ore ground to 40-60 mesh with a moisture content of 5-10% is subjected to acid leaching by the batch method. The consumption of acid, the duration of leaching and the necessity for heating depend on the character of the ore and the required degree of extraction. After acid treatment, the pulp usually contains 15-20% moisture. The leaching process is best carried out with the extraction, which leads to a considerable reduction in the consumption of acid and to a reduction in gas evolution in a number of cases. Leaching by the method of acid mixing with extraction is carried out in blade mixers with a countercurrent arrangement. When using nitric acid for the leaching it is possible to use extraction agents such as isoamyl acetate, dimethyl phthalate, dibutyl ether or 5-10% solution of TBP in kerosene, which gives high extraction of the metal with a minimum consumption of acids in the leaching.

Further extraction of the extraction agent from the ore cakes can be carried out by the following methods: 1) distillation with heating; 2) washing the cakes with organic solvents (kerosene, hexane) and 3) repulping with water.

The extraction agent saturated with uranium is subjected to reextraction, water being used to reextract uranium from ethers and esters of carboxylic acids; uranium can be reextracted from TBP by a 3-5% solution of sulfuric acid, a 5-10% solution of ammonium sulfate or by ammonium carbonate solutions.

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INTERACTION OF URANIUM HEXAFLUORIDE WITH AMMONIA

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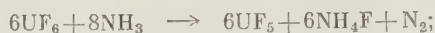
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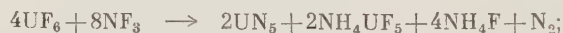
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The interaction of uranium hexafluoride with ammonia in the temperature range from -50 to +200°C has been studied. The total equations of the reactions are proposed:

1) for the temperature range -50 to -30°C



2) for the temperature range 0-25°C



3) for the temperature range 100-200°C



The rate of the reaction in the temperature range between -20 and +20°C has been evaluated. The thermal effect of the reaction in the temperature interval between -50 and -30°C varied between 50.8 and 83.6 kcal/mole, and, at -40°C it was in satisfactory agreement with the calculated value obtained from the proposed equation.

A whole series of papers was devoted to the interaction of uranium hexafluoride with various reducing agents. The reduction of uranium hexafluoride by hydrogen [1-3], by hydrogen chloride and bromide [4, 5], by carbon tetrachloride [6], by trichloroethylene [3], by ethylene and propane [7], etc., were studied. It has been pointed out [8] that uranium hexafluoride interacts with ammonia already at the temperature of dry ice (-72°C); in gas phase uranium hexafluoride is reduced by ammonia at a temperature of 300°C with formation of solid product containing 98% NH_4UF_5 and 2% of NH_4F . No data concerning the details of this problem are available.

In the present work we studied the interaction of uranium hexafluoride with ammonia in the temperature range from -50 to 200°C, with the purpose of determining the total equations of the reactions at different temperatures, and also the interaction rates of the reagents and the thermal effect of the reaction.

The interaction of solid uranium hexafluoride with liquid and gaseous ammonia was studied on an apparatus shown in Fig. 1.

Solid uranium hexafluoride (2.0-4.0 g), liquid ammonia in a quartz ampoule [0.01-0.5 g for the reaction $\text{UF}_6(\text{sol}) + \text{NH}_3(\text{g})$, and 0.5-2.0 g for the reaction $\text{UF}_6(\text{sol}) + \text{NH}_3(\text{l})$] were placed in a stainless steel bomb under thermostatzation. After the required temperature had been reached, the ampoule containing the ammonia was broken by a special device. The pressure of the system was controlled by means of a manometer. The solid and gaseous phases were analyzed. In some cases the reaction products were subjected to x-ray and thermal analysis.

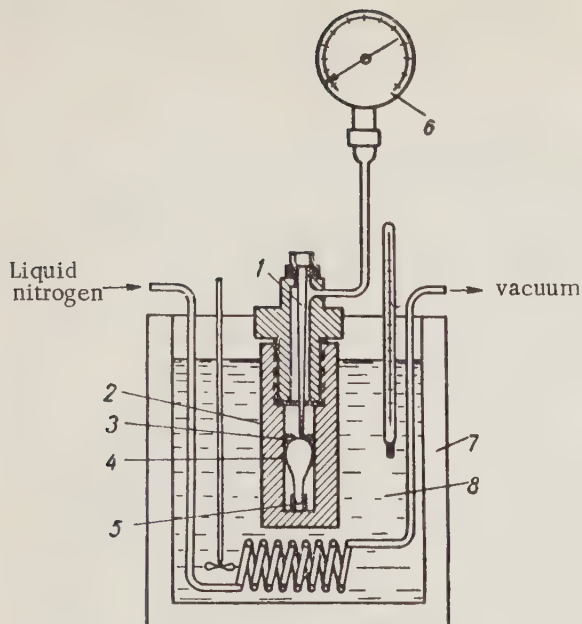


Fig. 1. Scheme of the apparatus for studying the interaction of solid uranium hexafluoride with liquid and gaseous ammonia: 1) breaking device; 2) reaction bomb; 3) solid uranium hexafluoride; 4) quartz ampoule with liquid ammonia; 5) base for the ampoule; 6) manometer; 7) Dewar vessel; 8) solution of the salts NH_4Cl , NaCl , ZnSO_4 , CaCl_2 , etc.

The interaction of uranium hexafluoride with ammonia in the gas phase was studied with the apparatus whose scheme is given in Fig. 2.

The reacting gases were previously heated at the given temperature; uranium hexafluoride was either diluted with argon or placed in the reactor without additions. In the first case the reaction proceeded more slowly, without clogging the connection of the apparatus and thus making the investigation simpler. The solid and gaseous products of the reaction were analyzed.

The thermal effect of the reaction between solid uranium hexafluoride and liquid or gaseous ammonia depending upon the temperature of the reaction was determined by a direct calorimetric method with a special apparatus (Fig. 3). The weight of uranium hexafluoride was 0.1-2g, the weight of ammonia was 0.01-0.2g for the reaction $\text{UF}_6(\text{sol}) + \text{NH}_3(\text{g})$, and 0.5-2g for the reaction $\text{UF}_6(\text{sol}) + \text{NH}_3(\text{l})$. The temperature of the calorimetric liquid (ethyl alcohol) increased by $0.2-1^\circ\text{C}$ at the expense of the heat of reaction.

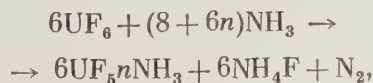
The investigations carried out showed that uranium hexafluoride, at temperatures between -50 and $+200^\circ\text{C}$ reacts with ammonia and forms a mixture of solid products. On the basis of visual observations it was found that in the temperature range from -50 to -30°C a white solid product was formed, which rapidly became green in air. At a temperature of -20 to 0°C a mixture of a white, a gray, and a green product was obtained; they took a stable green color by standing in the air. Finally, at temperatures

above 0°C , a green product was mainly obtained. However, it should be noted that in all cases, the composition of the solid phase included a white product becoming green in the air (uranium pentafluoride) when the experiments were done in an appropriate manner (rapid interruption of the reaction by cooling the reaction apparatus with liquid nitrogen). Already these observations permitted us to conclude that the reduction of uranium hexafluoride by ammonia is a complicated reaction, one of whose steps is the formation of uranium pentafluoride.

The results of the chemical analysis of the solid products obtained upon interaction of uranium hexafluoride with ammonia in the temperature range under study are given in Table 1.

As can be seen from these data, in the temperature range from -50 to -30°C a solid product having a roughly constant chemical composition is formed. According to the results of x-ray analysis, uranium tetrafluoride was not present in this product. It follows that in the reaction of uranium hexafluoride with ammonia in the temperature range from -50 to -30°C , only uranium pentafluoride was formed, and the fluorine ion separated forming with ammonia, ammonium fluoride. It should be noted that the ammonia in these reaction products was somewhat more than required by the stoichiometry of the formation of ammonium fluoride; the latter circumstance can be explained both by the formation of uranium pentafluoride ammoniates, and by the mechanical capture of ammonia.

The total reaction of interaction of uranium hexafluoride with ammonia in the temperature range from -50 to -30°C is described by the following equation:



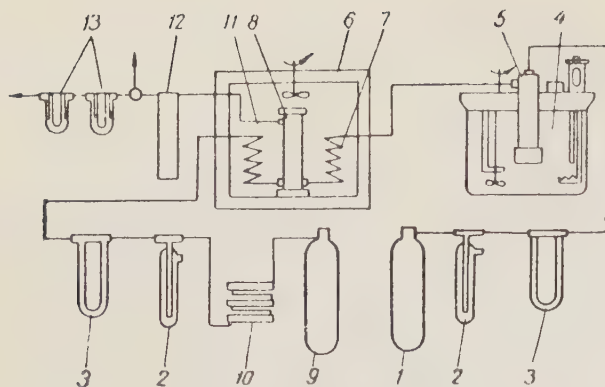


Fig. 2. Scheme of the apparatus for the study of the interaction of uranium hexafluoride with ammonia in gas phase: 1) Argon cylinder; 2) monostat; 3) flowmeter; 4) water thermostat; 5) uranium hexafluoride vaporizer; 6) air thermostat; 7) preliminary heater of uranium hexafluoride; 8) reactor; 9) ammonia cylinder; 10) drying columns with KOH; 11) preliminary heater of ammonia; 12) condenser; 13) trap.

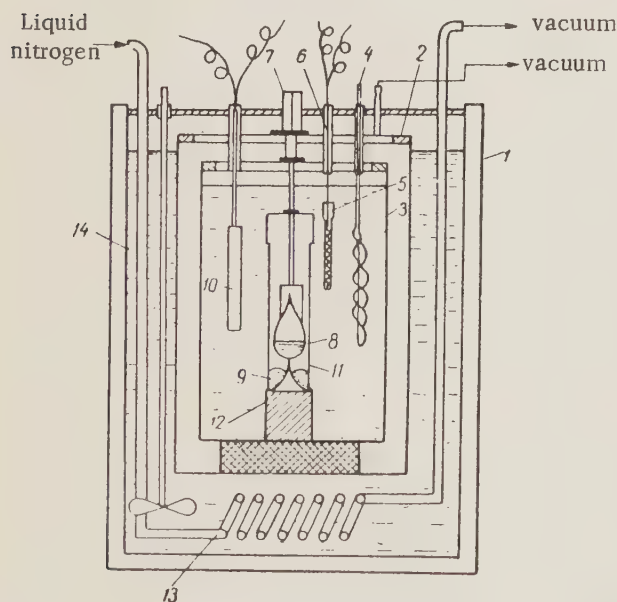


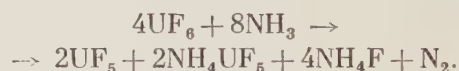
Fig. 3. Scheme of the calorimetric apparatus: 1) Dewar vessel; 2) external wall of the calorimeter; 3) internal wall of the calorimeter; 4) stirrer; 5) heater; 6) stoppers of a thermal insulator; 7) breaking device (of ebonite); 8) quartz ampoule with liquid ammonia; 9) solid uranium hexafluoride; 10) resistance thermometer; 11) reaction bomb; 12) thermal insulator; 13) liquid nitrogen vaporizer; 14) solution of the salts NH_4Cl , NaCl , ZnSO_4 , CaCl_2 , etc.

where $n = 0.73$. The calculated and experimental data (mean values in %) concerning the composition of the solid phase are reported below:

Data	U^{+4} *	U_{tot}	F	NH_3
Calculated	50.0	62.2	29.8	7.6
Experimental	50.6	61.2	31.0	7.6

*In all such results we show the percent contents of U^{+4} with respect to its total amount.

For higher temperatures (-20 to 0°C) the reduction reaction of uranium hexafluoride by ammonia did not stop at the formation of uranium pentafluoride; it proceeds further; in the reaction products uranium ammonium pentafluoride was detected (from data of differential-thermal analysis and x-ray structure analysis). In the temperature range from 0 to $+25^\circ\text{C}$, a product having a roughly constant chemical composition was formed; this product corresponded to a 50% reduction of the uranium pentafluoride formed to uranium ammonium pentafluoride. The total reaction is described by the following equation



The calculated and experimental data (mean values in %) concerning the composition of the solid phase are given below

Data	U^{+4}	U_{tot}	F	NH_3
Calculated	75.0	62.8	30.0	6.6
Experimental	77.6	63.1	29.9	6.7

For still higher temperatures (100 - 200°C) the uranium pentafluoride was completely reduced to uranium ammonium pentafluoride. The total reaction of reduction of uranium hexafluoride by ammonia in this temperature range is described by the equation



TABLE 1

Composition of the solid products of the reaction between uranium hexafluoride and ammonia.

Reaction temperature °C	Degree of reduction of uranium, %	Contents of the individual components in the reaction products, %		
		U	F	NH ₃
-50	50,9	61,3	30,5	7,8
-40	50,3	60,8	31,3	7,6
-30	50,6	61,6	30,4	7,7
-20	59,5	60,2	32,5	6,9
-15	64,4	58,7	33,9	7,2
-10	71,4	59,8	33,1	7,1
-5	73,8	60,0	32,8	6,8
0	77,5	62,9	30,2	6,7
+15	77,6	62,2	31,0	6,6
+25	77,7	63,2	29,7	6,7
+100	98,7	61,6	29,1	9,2
+150	99,1	62,0	29,3	8,9
+200	99,5	62,2	29,0	8,7

Note: The gas phase contained in all cases nitrogen; a quantitative analysis of the gas phase relating to the content of the nitrogen was carried out only for the reactions performed in the temperature range 100-200°C; the data obtained in these experiments agreed well with the calculated data referring to the contents of nitrogen computed from the equation of the high-temperature reaction between uranium hexafluoride and ammonia given below.

certain increase of the temperature within the bomb, due to the thermal effect of the reaction; therefore we did not carry out a mathematical elaboration of the results obtained. However, preliminary calculations showed that the order of the reaction decreased from 2 in the first 5-10 sec to 0.5 after 3-5 min. This circumstance was apparently due to surface effects (diffusion of ammonia, absorption of ammonia by solid uranium hexafluoride, and by the reaction products with subsequent interaction). The experiments carried out at temperatures of 100-200°C showed that uranium hexafluoride in these conditions interacted practically instantaneously with ammonia.

By direct calorimetric measurements we established that the reaction of uranium hexafluoride with ammonia took place with a large production of heat. The values of the thermal effect of this reaction in the temperature range from -50 to -30°C are shown in Table 2.

The thermal effect of the reaction calculated according to the equation of low temperature interaction of uranium hexafluoride with ammonia (at a temperature of -40°C), appears to be equal to 65 kcal/mole, in satisfactory agreement with the value found experimentally for the thermal effect of the reaction at this temperature. This circumstance supports again the correctness of the total equation proposed above for the reaction between uranium hexafluoride and ammonia at low temperatures.

The calculated and experimental data (average values in %) concerning the composition of the solid phase are reported below:

Data	U ⁺⁴	U _{tot}	F	NH ₃
Calculated	100,0	61,3	29,4	8,7
Experimental	99,1	61,9	29,1	8,9

The rate of the reaction between uranium hexafluoride and ammonia in the temperature range from -20 to +20°C was determined from the change of the pressure of ammonia in a closed volume. This could be done due to the fact that at low temperatures the reduction reaction of uranium hexafluoride by ammonia leads to the formation mainly of solid products; as to the nitrogen produced, its quantity was comparatively unimportant with respect to the amount of ammonia that had reacted, the more so because it could always be taken into account. Naturally it was also necessary to take into account the influence of the thermal effect on the change of the pressure of ammonia. The results of the determination of the rate of the reaction between solid uranium hexafluoride and gaseous ammonia are presented in Fig. 4. The values of the pressure changes for ammonia were calculated in moles according to Planck's [9] formula. As can be seen from Fig. 4, the rate of the reaction, as could be expected, decreased when the temperature decreased. Even at a temperature of -20°C the reaction stopped in 3-5 min.

In the experiments carried out it was not possible to keep strictly constant the surface of the solid uranium hexafluoride, and also to avoid a

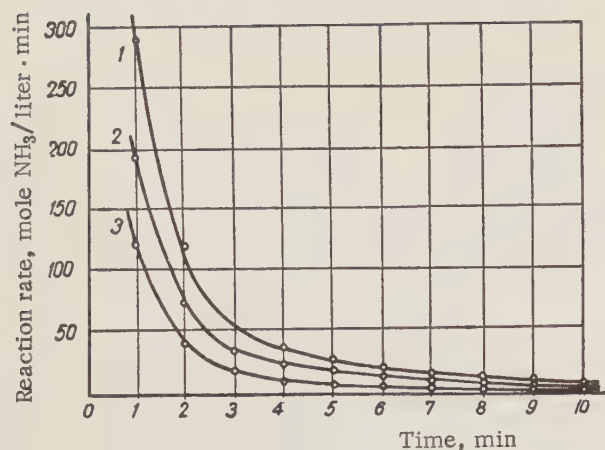


Fig. 4. Dependence of the rate of the reaction between ammonia and solid uranium hexafluoride upon time. Reaction temperature, °C: 1) -20; 2) 0; 3) +20.

starting with temperatures of -30 to -25°C; this fact was apparently related, first, to the increase of the heat of reaction due to the complete reduction of uranium pentafluoride, and, second, to the formation of uranium ammonium pentafluoride. The values of the thermal effects for this temperature range are being refined at present.

TABLE 2

Thermal effect of the reaction of interaction of solid uranium hexafluoride with liquid ammonia.

Reaction temperature, °C	Thermal effect of the reaction kcal/mole
-50	50,8±1,5
-45	59,1±1,8
-40	67,0±2,0
-35	75,4±2,3
-30	83,6±2,5

* Average of 3-5 experiments.

Preliminary experiments showed that the thermal effect of the reaction between solid uranium hexafluoride and gaseous ammonia increased rapidly,

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THE FLOCCULATION OF PULP AND POLYACRYLAMIDE-TYPE FLOCCULENTS

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The processes of dividing hydrometallurgical pulps into solid and liquid phases with subsequent washing out of the dispersed solid particles from the solute are very important in the technology of processing ores and concentrates of uranium, lithium, zirconium, and other metals. Flocculation of pulp, i.e., the formation of more or less large and strong aggregates of particles, is accomplished by special flocculating reagents. Their action facilitates dividing and washing out the solid phase from the solute. This article describes the properties and methods of obtaining the most effective flocculents and gives the results of an investigation of the flocculation processes of aqueous, acid, and carbonaceous hydrometallurgical pulps. The article is of interest for a wide circle of specialists working in industrial enterprises, in factory laboratories, and in research institutes studying the concentration of ores and the hydrometallurgy of uranium and other metals.

High-molecular weight or surface-active substances used as flocculents must satisfy the following basic requirements. The molecules of a flocculating reagent must contain active groups capable of being adsorbed on the surface of solid particles and must be linear and of definite length in order to form bridges between the particles. Flocculating reagents must be soluble in water; the reagents must be capable of reacting even when only very low concentrations of them are present in the pulp. The floccules must be strong enough to form porous cakes during filtration of the pulp and to retain as little liquid as possible during coagulation by settling.

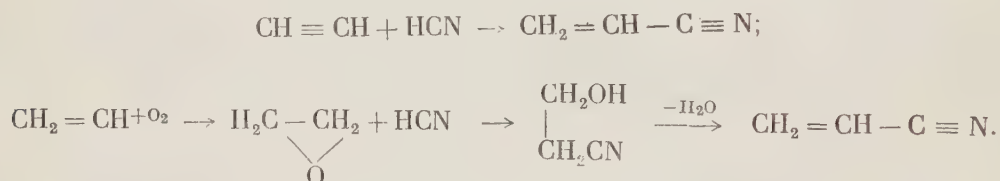
The intensity of the formation of floccules, their shape, size, and strength depend on the physicochemical properties of the suspension being flocculated and on the flocculating reagent. Here decisive importance is attached to the ratio of the liquid phase to the solid phase in suspension, the size and composition of the particles of the solid phase, the ion composition of the liquid phase, the concentration of the flocculent in the working solution introduced into the suspension and the method of mixing this solution with the suspension. The presence in the pulp of precipitates of metal hydroxides suppresses the action of the flocculents. On the other hand the effectiveness of their action increases in diluted pulps containing small amounts of electrolytes and also with a decrease in the concentration of the working solution of the flocculent. The latter requires the use of solutions with a concentration of 0.5-10 g/liter in practice. Circulating solutions can be used for the preparation and dilution of the flocculents in conformity with the technological pattern of the production cycle.

When introducing the flocculent into the pulp, it is necessary to see that it is rapidly and uniformly distributed throughout the entire mass of the pulp. However, any intensive mechanical mixing of the pulp, during which the floccules forming can be irreparably destroyed, is completely inadmissible. Under production conditions the most acceptable method is to introduce the flocculent into a stream of pulp flowing in a pipe, trough, or channel. In many cases the effectiveness of flocculation is increased if the flocculent is introduced into the pulp not all at once but in individual portions. In this case the pulp should be well mixed with the flocculent after the introduction of each portion.

The addition of inorganic compounds (lime, iron and aluminum salts, acids, alkalis) and organic substances (animal glue, starch, flour, etc.) has been used for many years to accelerate coagulation and filtration of pulps. We proposed new flocculents of natural origin: VL on a base of seaweed, KLZh on a base of linseed cake, M-42 on a base of potato pulp, etc. [1]. The flocculating power of these reagents is due to the fact that they contain high-molecular weight compounds belonging to the classes of polysaccharides and proteins. As a result of experimental investigations it was established that the new flocculents are more effective than the reagents previously used during coagulation of hydrometallurgical [2-4] and coal slime pulps [5].

In recent years synthetic flocculents which are high-molecular weight water-soluble substances have acquired increasing importance. In spite of the comparatively high cost of synthetic flocculents, their use is profitable, since in effectiveness of action they considerably excel all flocculents of natural origin [3-8]. A list of foreign high-molecular weight synthetic flocculents and starting materials for their production was published in [9]. In our country and abroad synthetic flocculents are produced under various names given to these reagents by the inventors of the methods producing them [1, 9-12]. The results of laboratory investigations [3, 5, 6] and the experience of industrial enterprises of the coal [7, 13, 14] and metallurgical branches of industry [15-17] make it possible to consider that the most effective and universal flocculents are reagents of the polyacrylamide type, among which we may cite Separan 2610, which is produced in the USA by the Dow Chemical Company, the polyacrylamide produced by the Institute of High-Molecular Weight Compounds of the Academy of Sciences, USSR and the Institute of Halurgy, and a polyacrylamide flocculent (AMF), which is produced by our method at one of the factories in the Gorkiy economic region.*

The starting material for the synthesis of the polyacrylamide-type compounds is acrylonitrile (AC). Several industrial methods for synthesizing acrylonitrile are known. The most widely used methods are those based on the reaction of acetylene or ethylene oxide with prussic acid according to the formulas



Other technological methods of synthesizing acrylonitrile and the kinetics of its polymerization are described in [18]. Acrylonitrile is easily polymerized and forms polyacrylonitrile (PAC), which can be converted into water-soluble compounds by hydrolysis with sulfuric acid or alkali solutions. The process of hydrolysis of PAC, depending on the temperature and the concentration of the sulfuric acid, leads to the formation of products of various chemical compositions. After hydrolysis of PAC with 75-95% cold sulfuric acid for 4 hrs, a water-soluble polymer is produced which contains imide and amide groups. The reaction of PAC with 65-95% acid when heated leads to the formation of polymers which dissolve only in dimethyl formamide. A reduction in the concentration of the sulfuric acid to 50% leads to an increase of polyacrylic acid in the hydrolysis products [19]. We also obtained water-soluble products by treating PAC with solutions of sodium hydroxide at 80-90°C for 10 hrs.

The products of the hydrolysis of PAC with sulfuric acid and an alkali can be used as flocculents. However, as our experiments showed, the effectiveness of these reagents is usually lower than the effectiveness of Separan 2610 and the flocculent AMF obtained by the polymerization of acrylamide.

Of the many known methods of producing acrylamide the most important are those based on hydrolysis of acrylonitrile in the presence of sulfuric acid and differing only in the method of separating the acrylamide by diluting the reaction mixture with ice water, by neutralizing with lime, filtering off the calcium sulfate, and concentrating the aqueous solution in a vacuum, by neutralizing the solution with ammonia, and many other methods of separation and polymerization of acrylamide [20-26].

*The method was developed together with engineers M. P. Vilyanskyi and N. P. Pashkin. Industrial application of the method was accomplished with the active participation of factory workers É. A. Kulev and R. Z. Khantsis.

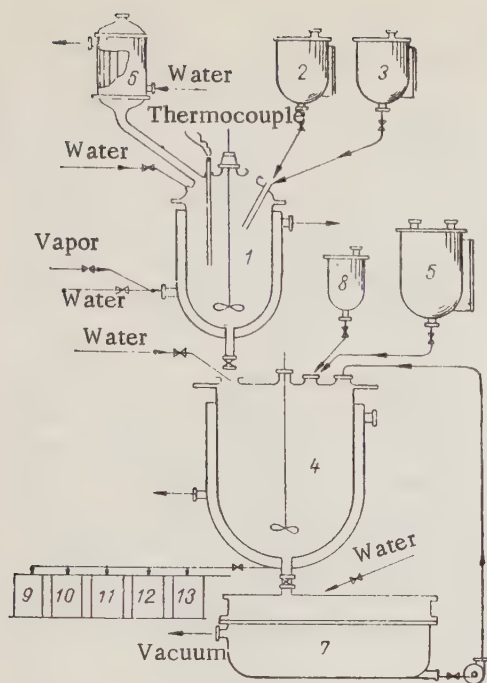


Fig. 1. Diagram of the apparatus for producing AMF flocculents: 1) reaction vessel; 2) service tank for sulfuric acid; 3) service tank for acrylonitrile; 4) reactor; 5) service tank for ammonia water; 6) reflux condenser; 7) vacuum filter; 8) service tank for ammonium persulfate; 9-13) polymerization chambers.

is not observed. The entire portion of the acrylonitrile was added to the vessel in one hour, after which the reaction mixture was held at 90°C for 45 min.

Neutralization of the solution after dilution with water can be accomplished by means of one of the following reagents: calcium carbonate, lime, sodium hydroxide, sodium carbonate, water and gaseous ammonia. As an initiator of polymerization we used ammonium persulfate activated by sodium sulfite. The solutions obtained after neutralization of the acrylamide sulfate by ammonia without separation of the ammonium sulfate were subjected to polymerization. The pH of the solution after polymerization fluctuated between 9 and 8.

The polymer forming under these conditions is a viscous, gelatinous mass which, during a more or less long time, depending on the thickness of the layer, the temperature, and the exchange ratio, dries in the air and is transformed into a light, brittle material containing from 35 to 50% active substances. The AMF flocculent obtained readily dissolves in water during intensive mixing and forms homogeneous solutions.

Comparative investigations of the effectiveness of the reagent AMF and other flocculents of natural and artificial origin were conducted on different kinds of slime pulps obtained by the metallurgical, coal, and chemical industries. It should be noted that the presence in the pulp of precipitates of metal hydroxides and colloidal silica considerably reduces the effectiveness of a flocculent's action.

The effect of additions of flocculents on the sedimentation of pulps was studied in measuring tanks. For this purpose the change in time of the height of the layer of clarified liquid was recorded. In order to keep the duration and the intensity of mixing the pulp with the flocculent solution constant, the tanks with the pulp were attached in a special stand in which they were simultaneously turned over in order to mix the pulp with the flocculent solution added to it.

The results of processing the sedimentation curves made it possible for us to obtain not only a qualitative but also a quantitative characteristic of the effectiveness of the flocculent's action in the form of values ex-

The hydrolysis of acrylonitrile with sulfuric acid with subsequent polymerization of the reaction products was the basis of the method of producing AMF. Synthesis of the reagent AMF was accomplished on an apparatus, the diagram of which is shown in Fig. 1. The reaction between the acrylonitrile and sulfuric acid proceeds very violently and is accompanied by considerable generation of heat. An increase in the temperature of the reaction mixture to 130°C can cause dangerous splashing and explosions of the reaction vessels. The possibility of dangerous complications arising at this stage is eliminated if the acrylonitrile is added gradually and uniformly to the reaction vessel containing the sulfuric acid and polymerization inhibitor (powdered copper, elementary sulfur), while the mixture is stirred intensively. The behavior of the process at a temperature of about 100°C enables the reaction to go to completion in 40-50 min.

The following optimum conditions for realizing the process were established by numerous experiments: 84.5% sulfuric acid for a calculation of 1 mole H_2SO_4 and 1 mole acrylonitrile was poured into the reaction vessel and elementary powdered sulfur or a solution of methylene blue added. The mixture was heated to 70-80°C and then the acrylonitrile was added to the reaction vessel slowly and uniformly with the mixer operating. As a result of the exothermal reaction the temperature of the mixture increased to 100°C. If cold water (with a temperature of 17-18°C) is circulated through the jacket surrounding the reaction vessel, then a further increase in the temperature

pressing the height of the layer of the clarified liquid for a definite interval of time from the start of the experiment, or in the form of an index of the requisite specific area of pulp coagulation.

Figure 2 contains curves describing the effect of the additions of various flocculents on the sedimentation rate of pulp obtained after sulfuric acid leaching of ore containing 35% quartz and 55% feldspar.* The ratio of the liquid phase to the solid phase in the initial pulp, which is the overflow of the classifiers, was 7.3 : 1; the pH of the solution was 1 : 1. The temperature of the experiments was 20°C. The solid phase of the pulp did not contain particles larger than 0.074 mm. In this series of experiments the expenditure of the flocculents was 100 g/m³. The shape of the curves did not change when the flocculents had other rates of expenditure. From a comparison of the results obtained it can be seen that the effectiveness of the AMF synthetic flocculents and Separan 2610 are approximately the same and considerably higher than flocculents of natural origin.

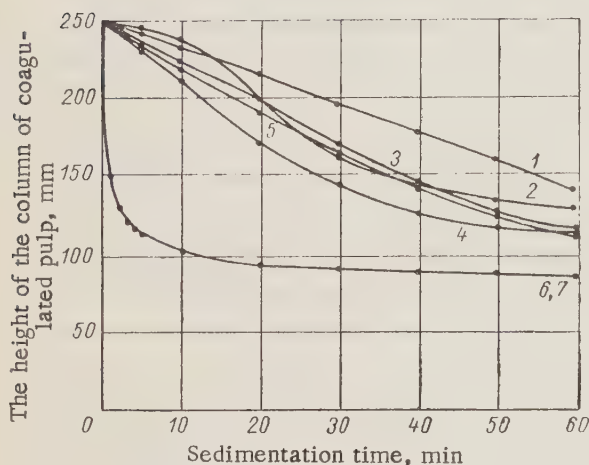


Fig. 2. The effect of additions of flocculents on the sedimentation rate of sulfuric acid pulp. The expenditure of the flocculents is 100 g/m³: 1) starting pulp; 2) carboxymethylcellulose; 3) M-42; 4) KLZh; 5) VL; 6) AMF; 7) Separan 2610.

The next series of experiments was made with aqueous pulp, which was the overflow of spiral classifiers operating in a closed cycle with ball mills at the pulverization stage. The ore material contained up to 70% clays (montmorillonite-kaolinite). The ratio of the liquid phase to the solid phase in the pulp was 6 : 1, the solid phase of the pulp did not contain particles larger than 0.3 mm. Figures 3 and 4 contain curves describing the sedimentation rate of the aqueous ore at various rates of expenditure of the flocculents. In this series of experiments the flocculent AMF was a somewhat more effective reagent than Separan 2610.

The flocculating power of the reagent AMF is manifested most clearly when it is introduced into pulps obtained after soda leaching of the metals from the ores. Figure 5 contains curves describing the process of the sedimentation of pulp obtained after soda leaching of ores containing about 25% carbonates and 65% aluminosilicates. The content of Na₂CO₃ in the liquid phase of the pulp was 25 g/liter; the ratio of the liquid phase to the solid phase was 3 : 1; the solid phase did not contain particles greater than 0.1 mm; the temperature of the pulp was 20°C. The experiments were made at various rates of expenditure of the AMF flocculents.

Figure 6 represents the results of the experiments showing the effect of the value of the ratio of the liquid phase to the solid phase in carbonate pulp on the sedimentation rate. These dependences were obtained while investigating an entire series of soda pulps obtained after leaching ores of various compositions. It was established that the flocculent AMF at a rate of 50-150 g/m³ very effectively flocculates not only diluted but also rather dense soda hydrometallurgical pulps containing up to 40% solid phase. In this case a high productivity of coagulation exceeding 10 tons per m² per day and the production of coagulated pulp with a ratio of the liquid to the solid phase of about 0.8 : 1 are assumed. These circumstances are especially important when organizing the technological process of treating the ores with the use of continuous counterflow decantation washing of the pulp after leaching with the product of the clarified metalliferous solutions for their further processing by extraction, sorption, or other methods.

The effectiveness of continuous counterflow decantation washing can be calculated by the following formula:

$$E = \frac{M^{n+1} - M}{M^{n+1} - 1} 100\%,$$

* V. N. Palagina performed the experiments.

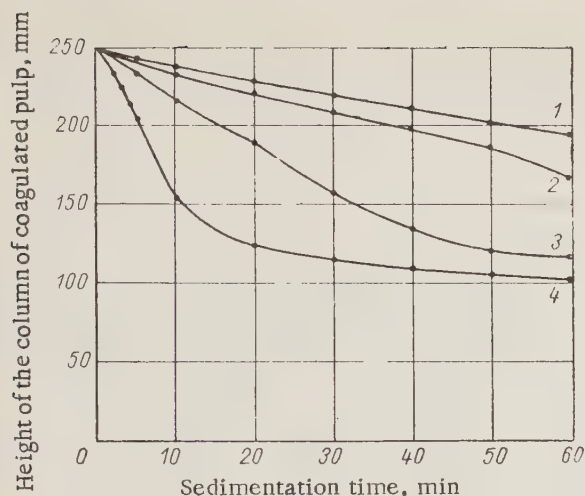


Fig. 3. The effect of additions of flocculents on the sedimentation rate of aqueous ore pulp. The expenditure of the flocculents is 20 g/m^3 : 1) Starting pulp; 2) seaweed grit BVLK; 3) Separan 2610; 4) AMF.

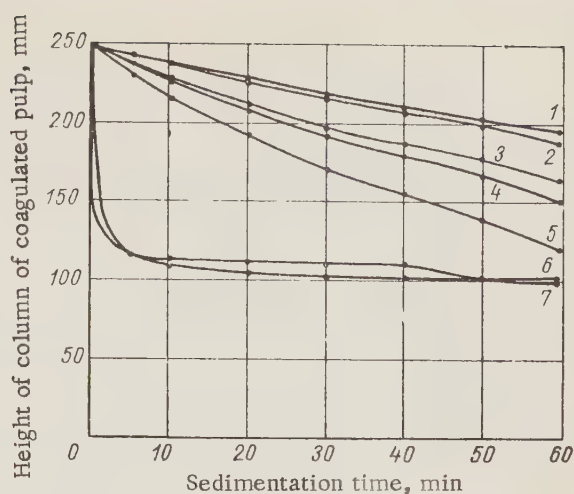


Fig. 4. The effect of addition of flocculents on the sedimentation rate of aqueous ore pulp. The expenditure of the flocculents is 50 g/m^3 : 1) starting pulp; 2) M-42; 3) KLZh; 4) VL; 5) BVLK; 6) Separan 2610; 7) AMF.

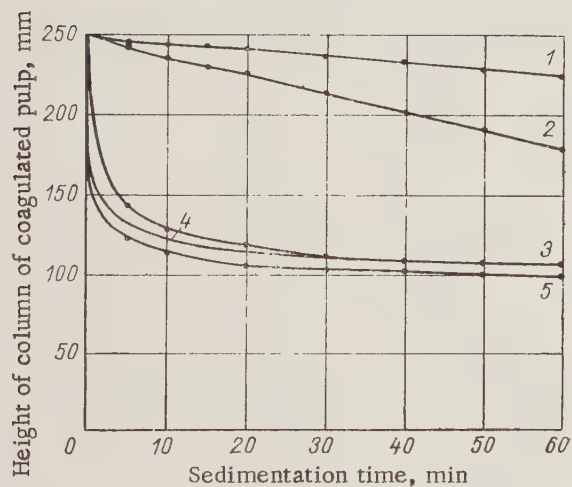


Fig. 5. The effect of the expenditure of the reagent AMF on the sedimentation time of the carbonate pulp: 1) starting pulp. Expenditure of AMF (g/m^3): 2) 20; 3) 50; 4) 100; 5) 200.

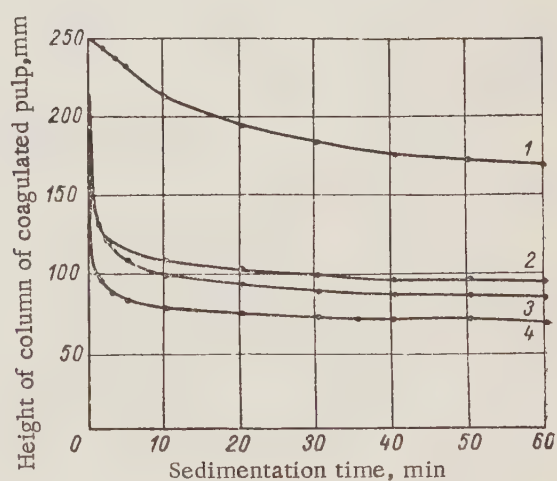


Fig. 6. The effect of the ratio of the liquid phase to the solid phase in carbonate pulp on the effectiveness of flocculation with the addition of 100 g/m^3 AMF. Ratio of the liquid phase to solid phase in initial pulp: 1) 1.7 : 1; 2) 3.1 : 1; 3) 4.4 : 1; 4) 5.0 : 1. Ratio of the liquid phase to solid phase after 24 hrs: 1) 0.7 : 1; 2) 0.91 : 1; 3) 0.94 : 1; 4) 0.96 : 1.

where E is the effectiveness of washing (%); M is the modulus of the washing expressing the ratio of the quantity of the liquid removed in overflow to the quantity of liquid remaining in the coagulated pulp at each stage of coagulation; $\underline{n}$ is the number of coagulation stages in the process. Figure 7 depicts this dependence graphically. It is seen from the figure that with one and the same number of coagulation stages the effectiveness of washing sharply increases with an increase in the modulus of washing or, what amounts to the same thing, with an increase in the density of the coagulated pulp.

Thus the use of the flocculent AMF can assure high technical-economic indexes of the process of continuous counterflow decantation washing of metal. In this case a comparatively small yield of commercial metalliferous solutions is assured per ton of ore when their concentration is sufficiently high.

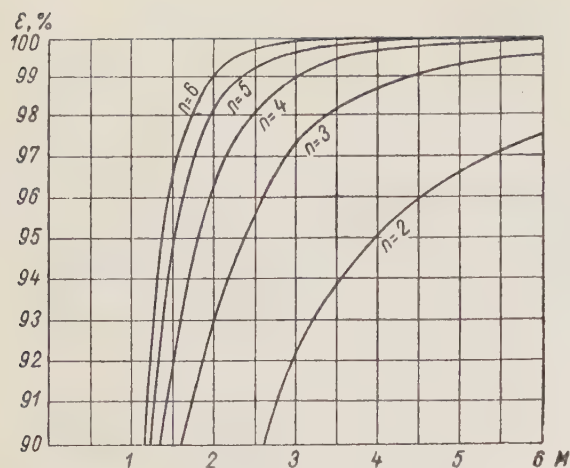


Fig. 7. The dependence of the effectiveness of continuous counterflow decantation washing on the value of the washing modulus M and the number of stages of washing n .

The introduction of flocculents into industry will make it possible to improve considerably the quantitative and qualitative indices of the process of coagulation and sedimentation of pulp.

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DETERMINATION OF ABSORBED DOSES IN ORGANISMS EXPOSED TO EMANATIONS AND THEIR DAUGHTER PRODUCTS

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Earlier, we considered the problem of absorbed doses created by short-lived α emitters in inhaling radon [3]. The present paper describes certain methods for determining absorbed doses due to radon itself, short-lived RaB and RaC β -emitters, and long-lived elements of the radon family. Similar calculations were performed for the thoron and actinon families, as a result of which new values for the maximum allowable concentration of thoron and actinon in air are recommended. The proposed method for determining absorbed doses can be used in the case where elements of any radioactive chain have penetrated the organism. It is demonstrated that personnel health monitoring with respect to the γ radiation of emanation daughter products that have settled in the respiratory system is a problem which can be solved in a practical manner.

The following peculiarities are encountered in determining the amount of absorbed energy when the organism is exposed to the radiation of emanations and their daughter products.

1. Due to the presence of a radioactive chain, the actual irradiation is due to the isotopes which directly penetrated the organism as well as those which are formed in the organism during the delay of each member of the radioactive family;

2. Among the isotopes participating in irradiation, there are α emitters as well as β and γ emitters;

3. These isotopes have different half-lives [from a few microseconds (RaC^{*}) to several tens of years (RaD)] and different biological half-lives;

4. The energy per single decay is very large (for radon and thoron, it is equal to ~ 20 Mev due to the α emitters alone);

5. All the emitters of the radon family can be divided into three fractions with respect to the relationship between the biological half-life and their half-life (λ_b and λ_h , respectively): a) radon, for which $\lambda_b \gg \lambda_h$ ($\lambda_{\text{eff}} \approx \lambda_b$), b) short-lived radon decay products (RaA, RaB, and RaC), for which, according to data from [1 and 2], $\lambda_b \ll \lambda_h$ ($\lambda_{\text{eff}} \approx \lambda_h$) in elimination from the respiratory system (exhaling) as well as in elimination from the organism as a whole, and c) long-lived radon decay products (RaD, RaE, and RaF), for which λ_h and λ_b are quantities of the same order ($\lambda_{\text{eff}} = \lambda_b + \lambda_h$);

6. All emanation isotopes enter the blood stream directly from the surrounding air and are rather quickly eliminated from the organism when the subject leaves the atmosphere containing emanations;

7. Short-lived daughter products of emanation decay are retained in the respiratory system, where they decay almost entirely before they are eliminated in appreciable quantities;

8. Long-lived radon decay daughter products are retained over a long period of time in some organs (especially in bones) and cause prolonged irradiation of the organism even after the penetration of radon has stopped.

Thus, the irradiation of organisms is a very complicated process in this case because of the presence of different emitter types (α , β , and γ) and also because the indicated fractions produce different types of irradiation in different organs. The presence of polonium isotopes in this family makes the exposure to emanations and their daughter products especially dangerous. It should be emphasized that according to the classification adopted at the International Conference of the Radiation Protection Experts of the International Organization of Labor (Geneva, November-December, 1959), Po^{210} as well as Ra^{226} and Ac^{227} with their daughter products are classed as the group of isotopes with the highest toxicity.

The expressions for calculating absorbed doses due to short-lived radon daughter products emitting α radiation have been derived earlier [3].

For calculating the amount of absorbed energy due to radon itself, the following must be borne in mind. From the over-all number of radon atoms which have penetrated the organism, only a small number of radon atoms decay while dwelling in the organism (for radon, the value of λ_b is considerably larger than the value of λ_h). For instance, in our experiments where radon water was administered to rats, it appeared that out of 1000 radon atoms introduced in the organism only ~ 2.5 atoms decayed inside the organism. Experiments where rats inhaled radon and were bathed in radon baths yielded λ_b values close to those obtained in administering radon water [4].

If we denote the number of radon atoms which decayed inside the organism by $\underline{n}$, we shall obtain the magnitude of the absorbed energy due to radon and its daughter products which accumulated as a result of the presence of radon in the organism:

for α emitters,

$$D'_{\text{Rn}} = n(E_{\text{Rn}} + E_A + E_{C'}); \quad (1)$$

for β emitters,

$$D''_{\text{Rn}} = n(E_B + E_C)^*. \quad (2)$$

The number of decay events $\underline{n}$ can be analytically expressed in terms of the quantity which is measured directly in experiments, namely, the rate at which radon is exhaled.

From the curves in Fig. 1, which represent the accumulation of radon in the organism (rats) due to radon inhalation and exposure in radon baths (in the latter case, radon enters the blood through the skin), it is obvious that $\underline{n}$, which is numerically equal to the area enclosed by the curve, can be represented in the form of three elementary integrals. The first integral takes into account the accumulation of radon in the organism (this portion can be considered as linear up to the instant of time t_1). During the next interval of time, from t_1 to t_2 (removal of the subject from the "emanatorium" **), the radon content in the organism remains almost constant ($\underline{a}$ is the maximum amount of radon in the organism for unit radon concentration in the air to be inhaled). Finally, the third integral takes into account the number of radon decay events after the subject is removed from the emanatorium up to the instant of time t_3 , when the amount of radon in the organism is almost equal to zero. For this portion of the curve, the radon constant in the organism is proportional to $e^{-\lambda_b t}$.

By using the expression from [5], which relates the isotope activity in the organism to the isotope elimination rate, we obtain:

$$\left(\frac{da}{dt} \right)_{t_2} = \lambda_b a,$$

* E_{Rn} , E_A , and $E_{C'}$ are the energies of Rn, RaA, and RaC' α particles, respectively; E_B and E_C are the average energies of RaB and RaC β particles, respectively, which are equal to $\sim 1/3 E_{\text{max}}$. Generally speaking, the consideration of β radiation is necessary in connection with the fact that portions of tissue inaccessible to α particles will be irradiated by β particles.

**The term "emanatorium" designates a room where the air contains emanations.

where $(da/dt)_{t_2}$ is the rate of radon eliminations from the organism reduced to the time t_2 when the exposure ends.

Finally, we obtain:

$$D_{Rn} = (E_{Rn} + E_A + E_B + E_C + E_{C'}) q \frac{1}{\lambda_b} \times \\ \times \left(\frac{da}{dt} \right)_{t_2} \left\{ t_2 - \frac{t_1}{2} - \frac{1}{\lambda_b} [e^{-\lambda_b(t_3-t_2)} - 1] \right\}, \quad (3)$$

where $D_{Rn} = D'_{Rn} + D''_{Rn}$ is the integral absorbed dose due to radon and its daughter products which accumulated during the decay of radon in the organism. In Eq. (3), D_{Rn} is expressed directly in terms of the radon exhaling rate at the moment t_2 .

In our experiments on rats, for

$$q = 1.35 \times 10^{-5} \text{ C/liter}, \\ (da/dt)_{t_2} = 30.5 \times 10^8 \text{ decay} \times \text{liter/C} \times \text{min}^2, \\ \lambda_b \cong 0.023 \text{ min}, \text{ and } t = 2 \text{ hr}$$

the absorbed energy D'_{Rn} due to radon was approximately equal to 6250 erg.

If we consider that the value of $(da/dt)_{t_2}$ for humans is approximately 30 to 40 times as large as that for rats (the breathing rate of rats is ~ 0.5 liter/min, and the breathing rate of humans is ~ 20 liter/min) and if we assume that λ_b has the value indicated above, the absorbed dose (D_{Rn}/m) will be equal to ~ 0.25 mrad/yr for $q = 10^{-10}$ C/liter, $t = 6$ hr, and $m = 70$ kg. Equation (3) can be used without modifications for thoron and actinon. The total energy of α emitters for thoron (per single decay) differs little from the corresponding value for the radon family, and λ_b and $(da/dt)_{t_2}$ obviously must be equal for all emanation isotopes.

By using the method for short-lived α emitters, which has been described in [3], we obtained expressions for calculating the amount of absorbed energy due to short-lived RaB and RaC β emitters.

As in the case of short-lived daughter products emitting α radiation, which has been considered in [3], the integral absorbed dose due to β emitters can be calculated according to the equation

$$D_{sh}^{\beta} = aqv_t \delta \{ \eta_A [\varphi_A^{\beta}(t) + \Psi_A^{\beta}(t)] + \\ + \eta_B [\varphi_B^{\beta}(t) + \Psi_B^{\beta}(t)] + \eta_C [\varphi_C^{\beta}(t) + \Psi_C^{\beta}(t)] \}^*$$

or, if the η_B/η_A and η_C/η_A ratios are known, with respect to the value of A_{γ} (the amount of the γ radiation emitting RaC isotope which settled in the respiratory ducts):

$$D_{sh}^{\beta} = A_{\gamma} \frac{\eta_A [\varphi_A^{\beta}(t) + \Psi_A^{\beta}(t)] + \eta_B [\varphi_B^{\beta}(t) + \Psi_B^{\beta}(t)] + \eta_C [\varphi_C^{\beta}(t) + \Psi_C^{\beta}(t)]}{\eta_A \xi_A(t) + \eta_B \xi_B(t) + \eta_C \xi_C(t)}.$$

if the radon concentration q , the degrees of equilibrium η_A , η_B , and η_C , the delay coefficient δ , and the volume v_t inhaled per minute are known.

Figure 2 shows the graphs of the functions $\varphi_A^{\beta}(t)$, $\varphi_B^{\beta}(t)$, $\varphi_C^{\beta}(t)$, $\Psi_A^{\beta}(t)$, $\Psi_B^{\beta}(t)$, $\Psi_C^{\beta}(t)$, which are proportional to the contribution of RaA, RaB, and RaC β radiation to the over-all amount of energy absorbed during the stay in the emanatorium (φ_A^{β} , φ_B^{β} , and φ_C^{β}) and after leaving it (Ψ_A^{β} , Ψ_B^{β} , Ψ_C^{β}), respectively. From a comparison between the φ^{β} and Ψ^{β} functions [3], the contribution of β radiation to the total amount of energy absorbed is lower than the contribution of α rays by approximately one order of magnitude.

* The β index in the φ and Ψ functions indicates that the dose functions φ and Ψ pertain to β emitters. The absence of this index in the φ and Ψ functions in [3] indicates that these functions pertain to α emitters.

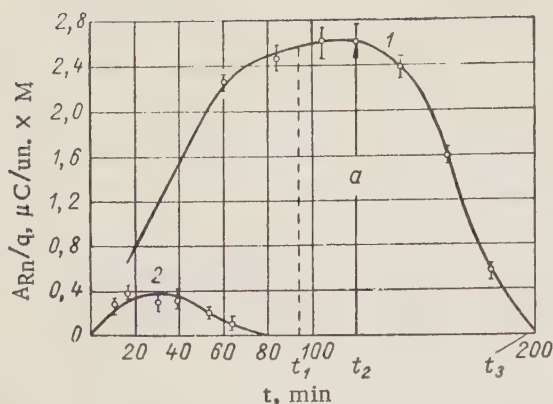


Fig. 1. Radon content in the organism (rats) in inhaling radon (curve 1) and during exposure in radon baths (curve 2).

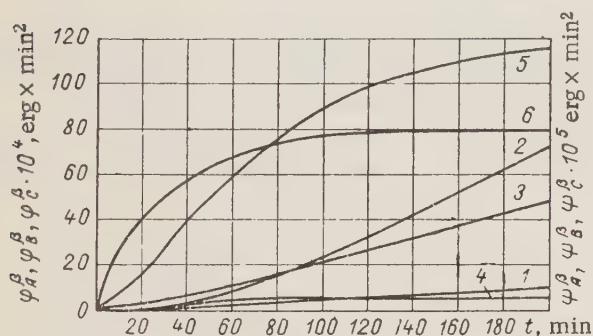


Fig. 2. Diagram of the φ^β and Ψ^β functions, which are proportional to the contribution of RaA, RaB, and RaC β radiation to the total amount of energy absorbed. 1) φ_A^β ; 2) φ_B^β ; 3) φ_C^β ; 4) Ψ_A^β ; 5) Ψ_B^β ; 6) Ψ_C^β .

bones will be $10^{-9} \times 3.7 \times 10^{10} = 37$ decays/sec. If we take into account the selective character of radium accumulation in bones, it is to be expected that the number of γ quanta per second due to radium will be considerably smaller in the portion of the body under investigation. The separation of RaC γ radiation created by short-lived radon decay daughter products from RaC γ radiation due to radium lodged in the organism can also be performed, since, in the first case, the intensity of γ radiation will decrease according to a certain law with an "average half-life" of ~ 34 min. All the above-mentioned justifies us in stating that devices for measuring the γ radiation of the human body, which have been described in literature, make it possible to measure the value of A_γ and thereby to calculate the absorbed doses with respect to the available graphs of the $\xi(t)$, $\varphi(t)$, and $\Psi(t)$ functions even for the cases where the content of radon decay daughter products in air is relatively small. Thus, personnel health monitoring seems to be feasible in plants where the air breathed by the personnel contains radon; this could be done by measuring the γ radiation of daughter products settled in the respiratory system and by subsequently calculating the absorbed dose with respect to α and β radiation. The absorbed dose due to radon can be determined on the basis of Eq. (3).

In our experiments [4], for $q = 1.35 \times 10^{-5}$ C/liter and $\eta = 0.2$, the integral dose due to α emitters absorbed by the respiratory system was

$$D_{sh} \simeq 18000 \text{ erg.}$$

In connection with the possibility of calculating the magnitude of the absorbed dose with respect to γ radiation, it is desirable to determine whether A_γ can be measured at all under actual conditions in plants where radon enters the air. For this, it is necessary to establish whether the γ radiation of short-lived radon daughter products which penetrated the respiratory system can be measured against the background of the γ radiation of isotopes contained in the human body (in the first place, K^{40} and Ra^{226}).

According to data from [6], the average potassium content in the organism is 0.215%, the specific γ activity of K^{40} is 3.5 γ quanta/sec.g, and the average energy of K^{40} γ rays is 1.46 Mev. Then, the over-all amount of K^{40} γ rays emitted from the entire body ($m = 7 \times 10^4$ g) will be $7 \times 10^4 \times 215 \times 10^{-5} \times 3.5$ γ quanta/sec $\simeq 525$ γ quanta/sec.

Considering that the measurement of short-lived daughter products with respect to γ radiation can be performed for a portion roughly consisting of 1/10 of the over-all body mass (the respiratory system and the adjacent organs) and assuming, according to data from [6], that the distribution of potassium in the organism is uniform, we find that the over-all number of decay events per second for this portion of the body is equal to ~ 50 . For $q = 10^{-10}$ C/liter, $\eta = 0.2$, $\delta = 0.5$, $v_t \simeq 20$ liter/min, and $t \simeq 3$ hr, A_γ will be equal to 500 decays/sec. Moreover, discrimination is always feasible in practice, since the average energy of RaB and RaC γ radiation is equal to 0.8 Mev and the average energy of K^{40} γ radiation is equal to ~ 1.46 Mev. The latest data on the radium content in human bones specify an amount not exceeding 10^{-9} g; consequently, for the entire body, the activity due to radium contained in

Thus, the absorbed dose created by short-lived products in the respiratory system proved to be much larger than the dose due to radon for the entire organism. In humans, for $v_t \approx 20$ liter/min, $\eta = 1$, $\delta = 0.5$, a weight of the respiratory organs of 0.8 kg, $q = 10^{-10}$ C/liter, and $t = 6$ hr, the absorbed dose is 1800 mrad/yr, i.e., it is larger than the dose due to radon itself by four orders of magnitude.

In our experiments on rats, besides dosimetric investigations, we performed parallel biological investigations, the results of which are in good agreement with these data. Drastic changes in the respiratory systems of the animals were observed, and they died one to three weeks after the beginning of irradiation [4].

Data in literature [6] on the spreading of various diseases of the respiratory system (including lung cancer) among miners in uranium mines indicate that the injurious influence is mainly due to the presence of short-lived radon decay daughter products in these mines (especially if poor ventilation conditions, which were formerly prevalent, are taken into account).

Similar relationships between the absorbed energy magnitude (with respect to α and β radiation) and the activity of decay products with respect to γ radiation were obtained for thoron and actinon families. The characteristics of thoron and actinon families are the following: the absence of a group of long-lived elements, short thoron and actinon half-lives (54.5 and 3.9 sec, respectively) in comparison with that of radon (3.8 days), and also, the presence of a relatively long-lived isotope in the thoron family (ThB). The amount of daughter products (A_γ) can be measured with respect to ThC'' γ radiation.

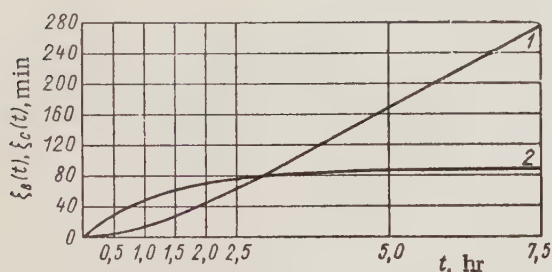


Fig. 3. Graphs of the $\xi_B(t)$ (curve 1) and $\xi_C(t)$ (curve 2) functions, which are proportional to the contributions of ThB and ThC to the value of A_γ (thoron family).

Figure 3 shows the graphs of the $\xi_B(t)$ and $\xi_C(t)$ functions, representing the contribution of ThB and ThC to A_γ , respectively (the contribution by ThA can be neglected). As in the case of the radon family, it is taken that the measurements of the γ radiation of thoron decay products which have settled in the respiratory system (A_γ) are performed with respect to the γ radiation of a preparation of known activity (mesothorium I in this case).

A comparison between the $\xi_B(t)$ values for radon and thoron indicates that, for $t \approx 3$ hr, $\xi_B(t)$ for thoron is more than twice as large as for radon; the increase in $\xi_B(t)$ takes place up to $t \approx 50$ hr.

Figures 4 and 5 show the φ^α and Ψ^α dose functions, which are proportional to the contributions of ThB and ThC to the value of the absorbed α radiation dose during the stay in a thoron atmosphere (function φ) and after leaving this atmosphere (function Ψ), respectively. Similar calculations were performed for β emitters.

From a comparison between the φ and Ψ functions for radon and thoron, it is obvious that, in the case of thoron, the irradiation is intensified after the subject is removed from the atmosphere containing thoron (the irradiation was performed during nonworking hours).

In the case of thoron, for $q = 10^{-10}$ C/liter, $\eta_B = \eta_C = 0.2$, $\delta = 0.5$, $v_t = 20$ liter/min, and $t = 6$ hr, the average dose due to the thoron daughter products absorbed by the respiratory system will be 9.3 mrad/day, which is approximately 20 times as large as the corresponding value for radon.* This fact provides reasons for reducing

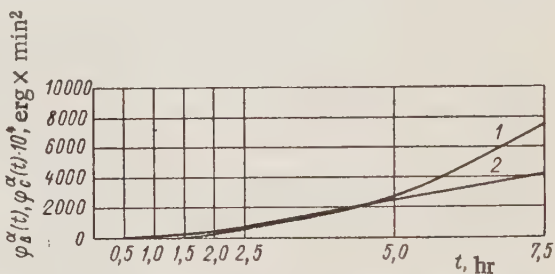


Fig. 4. Graphs of the $\varphi_B^\alpha(t)$ (curve 1) and $\varphi_C^\alpha(t)$ (curve 2) functions, which are proportional to the contributions of ThB and ThC to the absorbed α radiation dose (during stay in the emanatorium).

*The results of this calculation can be different if we take into account the biological elimination mechanism, which has not yet been sufficiently studied. In the general case, in the expressions for the ξ , φ , and Ψ functions, λ should be understood as $\lambda_{\text{eff}} = \lambda_b + \lambda_h$.

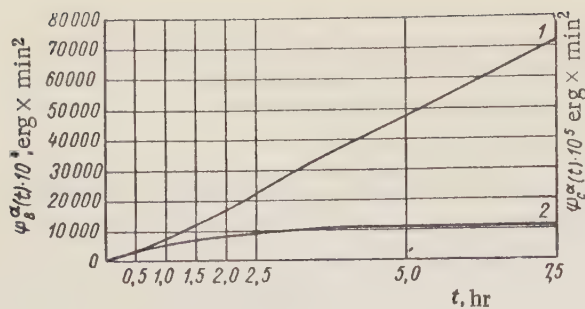


Fig. 5. Graphs of the $\Psi_B^\alpha(t)$ (curve 1) and $\Psi_C^\alpha(t)$ (curve 2) functions, which are proportional to the contributions of ThB and ThC to the magnitude of the absorbed α radiation dose (after removal from the emanatorium).

the maximum allowable concentration of thoron daughter products to 5×10^{-12} C/liter. The necessity of lowering the maximum allowable concentration of thoron daughter products was also indicated in [7].

The absorbed dose due to β emitters in inhaling thoron also considerably exceeds the corresponding value for the radon family. Thus, the ratio of Ψ_B^β for the thoron family to the analogous quantity for the radon family is approximately equal to 300. As in the case of radon, the absorbed dose is related to $A\gamma$ by a simple relationship.

The φ and Ψ functions of the actinon family were calculated. On the basis of calculations for actinon and its daughter products, a maximum allowable concentration of 5×10^{-11} C/liter can be recommended.

The accumulation of long-lived radon decay daughter products in the organism occurs in three ways:

1) The decay products enter the organism with the air inhaled (or with the water taken in); the number of RaD atoms will be denoted by N_D ;

2) RaD is formed from the radon which has decayed in the organism (N_D^*);

3) RaD is formed from short-lived products which have entered the organism from air (or water) (N_D^*).

The over-all amount of absorbed energy due to RaE and RaF can be calculated according to the quantities

$$D_E = E_E \lambda_D N_D \left(\frac{1 - e^{-\lambda_D t}}{\lambda_D} - \frac{1 - e^{-\lambda_E t}}{\lambda_E} \right);$$

$$D_F = \frac{\lambda_D \lambda_F N_D I_F}{(\lambda_F - \lambda_D)(\lambda_F - \lambda_E)} \left\{ (\lambda_F - \lambda_E) \frac{1 - e^{-\lambda_D t}}{\lambda_D} - (\lambda_F - \lambda_D) \frac{1 - e^{-\lambda_E t}}{\lambda_E} + \lambda_E \frac{1 - e^{-\lambda_F t}}{\lambda_F} \right\},$$

where $N_D = N_D^* + N_D^{**} + N_D^{***}$, and λ_D , λ_E , and λ_F denote the effective values; $\lambda_D^{\text{eff}} = 1.25 \times 10^{-8} \text{ sec}^{-1}$; $\lambda_E^{\text{eff}} = 4.31 \times 10^{-6} \text{ sec}^{-1}$; $\lambda_F^{\text{eff}} = 13.8 \times 10^{-8} \text{ sec}^{-1}$.

Calculations showed that internal administration of radon water with a few hundred microcuries, which was stored over a period of several days (which is always the case in practice), leads to the accumulation of RaD amounts in the organism that are very close to the maximum allowable amount (0.2 μC).

We did not measure the value of N_D^* in our experiments. If, according to the existing norms, we assume that the maximum allowable concentration of RaD in air is equal to 8×10^{-14} C/liter and that the portion of the isotope which penetrated the critical organ by inhaling is $f_{\text{air}} = 0.004$ [8], the absorbed dose due to RaF only will be ~ 20 mrad for $q = 10^{-10}$ C/liter, $\eta = 1$, $\delta = 0.5$, a weight of the critical organ (bone) of 7×10^3 g, and $t = 6$ hr, which considerably exceeds the dose due to radon.

The method that we applied for emanations and their daughter products can be used for any radioactive chain consisting of n members (in particular, if fission fragments penetrate the organism). For this the values of the decay constants $\lambda_1, \lambda_2, \dots$ and of the energies $E_1, E_2, \dots$ for the elements of the chain to be calculated are substituted for $\lambda_A, \lambda_B, \dots$ and $E_A, E_B, \dots$ in the obtained expressions for the ξ , φ , and Ψ functions.

From the above, the following conclusions can be drawn.

1. A relationship between the magnitude of the absorbed dose due to radon and the quantity which is usually measured under conditions in practice — the rate of radon elimination from the organism — has been obtained.

2. In addition to the calculations performed in [3] for short-lived α emitters, the φ^β and Ψ^β functions, which provide an idea of the contribution of RaB and RaC β emitters to the irradiation of respiratory organs in inhaling radon, have been calculated.

3. Calculations according to the obtained equations indicate that, in the case of equilibrium between the short-lived daughter products and radon in air, the absorbed dose due to α emitters in the respiratory system is larger than the dose due to radon in the entire organism by approximately four orders of magnitude.

4. It was shown that, for RaB and RaC concentrations in air exceeding $(0.1-0.2) \times 10^{-10}$ C/liter, the radiation (A_γ) of radon daughter products which settled in the respiratory system during an exposure time of ~ 1 hr and over can be calculated directly, which makes it possible to determine the absorbed α and β radiation energy by using devices for measuring γ radiation emitted by the human body.

5. The performed calculations show that the value of the maximum allowable concentration of thoron and its daughter products in air must be so reduced that it constitutes approximately one twentieth of the maximum allowable concentration of radon. For the actinon family, the corresponding magnitude must be $\sim 5 \times 10^{-11}$ C/liter.

6. The proposed method for calculating absorbed doses can be used in cases where any radioactive chain has entered the organism.

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LETTERS TO THE EDITOR

ABSORPTION SECTION OF FAST NEUTRONS

T. S. Belanova

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The method of neutron acceleration with a direct spherical geometry of the experiment was used in order to measure the absorption section of neutrons [1, 2]. Three neutron sources were used [Sb-Be, Na-D₂O, and Na-Be) giving neutrons with average energies 24 ± 3 , 220 ± 20 and 830 ± 40 kev respectively. As a neutron detector we used a modified long counter, having the same sensitivity (in $\pm 3\%$ limits) for neutrons of various energies [3, 4].

The substances studied were either in the form of cast metal or in the form of powders. The following corrections were introduced: 1) a correction for the scattering of neutrons on the walls of the containers; 2) a correction for the end dimensions of the neutron sources; 3) corrections related to the end distances between the detector and the neutron source; 4) a correction for multiple scattering of neutrons in absorbers, which was introduced in calculating the mean path of the neutrons in the absorbers according to the method described [1, 2].

TABLE

Absorption section (mbarn)

Elements	Energy, kev		
	24	220	830
Si	42 ± 8	10 ± 5	20 ± 10
S*	< 22	< 19	< 21
Ti	23 ± 6	21 ± 20	11 ± 20
Cr*	162 ± 7	—	—
Fe*	—	8 ± 2	—
Sr	99 ± 9	124 ± 30	105 ± 20
Mo	262 ± 11	70 ± 6	110 ± 10
Ag	984 ± 49	452 ± 40	230 ± 23
Te	240 ± 30	—	—
I*	990 ± 40	—	—
W*	—	—	91 ± 35
Au	605 ± 30	330 ± 30	224 ± 24
Hg*	230 ± 10	126 ± 6	103 ± 11
Th*	—	213 ± 5	—
U*	568 ± 10	146 ± 16	161 ± 6

*The values of absorption sections of these elements were refined.

The measurements of the absorption sections in gold and silver were carried out in thin-walled spheres (1 cm thick). The calculations of the absorption sections in gold and silver were carried out by Sh. S. Nikolaishvili. The results of the measurements are presented in the Table: the latter contains also the absorption sections of some elements measured previously [1]. The values of these sections were refined by new data concerning total and transport sections [5, 6].

The data of the present paper, with the exclusion of certain values, are in good agreement with the absorption sections reported in the papers [2, 7-10].

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CONVERGENCE OF THE SERIES IN THE MANY-VELOCITIES THEORY OF NEUTRON DIFFUSION

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The essential shortcoming in the many-group description of neutron thermalization [1] is the poor convergence of the series in associated LaGuerre polynomials if the source neutrons have energies greatly exceeding $k\theta$ (θ being the temperature of the retarder). In this event it is convenient to take as sources comparatively slow neutrons, the distribution of which is calculated in an increasing approximation. In certain problems the convergence of the above-indicated series may be improved without resorting to analysis in the increasing approximation. To do this, the solution of the stationary equation for the chaotic flow of neutrons at the point $\underline{r}$, viz., $N(\underline{r}, x)$ ($E = xk\theta$ being the neutron energy), is sought in the form

$$N(\underline{r}, x) = \psi(\underline{r}, x) + \varphi(\underline{r}, x), \quad (1)$$

where $\psi(\underline{r}, x)$ coincides with $N(\underline{r}, x)$ in the region $E \gg k\theta$. Then the density of neutron sources in this region in the equation satisfying $\varphi(\underline{r}, x)$ will be equal to zero. Let us expand $\varphi(\underline{r}, x)$ into a series of LaGuerre polynomials [1]:

$$\varphi(\underline{r}, x) = xe^{-x} \sum_{i=0}^{\infty} \frac{1}{i!(i+1)!} L_{(i)}^{(1)}(x) n_i(\underline{r}). \quad (2)$$

The convergence of the series (2) will be considerably better than the convergence of a series representing $N(\underline{r}, x)$. We shall consider two examples of the problem of finding a distribution function for neutrons in a heavy (mass number of the retarder nuclei equal to $M \gg 1$) monoatomic gaseous retarder.

1. Let us assume that the retarding medium is in the form of a collection of alternating plates of thickness $2a$ and $2b$ with uniformly distributed sources of monoenergetic neutrons, but with nonuniform absorption: The plates with thickness $2a$ contain a uniform mixture of retarder nuclei and nuclei that resonate in the super-thermal region of neutron energies, while the plates with thickness $2b$ are made up of the pure retarder. It is assumed that the presence of resonance-absorbent nuclei does not change the retarding properties of the medium. Then $\psi(\underline{r}, x)$ represents the spectrum of neutrons in the retarder with a constant lifetime T^* [2] (T^* corresponds to absorption by the retarder nuclei only), which satisfies the equation

$$\frac{2}{M\lambda_s} \left(x \frac{d^2 \Psi(x)}{dx^2} + x \frac{d\Psi(x)}{dx} + \Psi(x) \right) - \frac{\Psi(x)}{\sqrt{x}T^*} + Q(x) = 0. \quad (3)$$

Here λ_s is the mean free path relative to scattering, $Q(x) = A\delta(x - x_0)$ is the neutron source density. Assuming that $\frac{\lambda_s}{\sqrt{x}T^*} \ll 1$, in each plate we can write for the coefficients of the expansion $n_i(\underline{z})$ (2) a system of equations

in the two-group approximation ($i = 0, 1$):

$$\left. \begin{aligned} \frac{d^2 n_0(z)}{dz^2} - \frac{3}{\lambda_s} \left(\frac{1}{V x T(x)} \right)_{00} n_0(z) - \\ - \frac{3}{\lambda_s} \left(\frac{1}{V x T(x)} \right)_{01} n_1(z) = q_0, \\ \frac{d^2 n_1(z)}{dz^2} - \frac{3}{\lambda_s} \left(\frac{1}{V x T(x)} \right)_{10} n_0(z) - \\ - \left[\frac{3}{\lambda_s} \left(\frac{1}{V x T(x)_{11}} \right) + \frac{6}{M \lambda_s^2} \right] n_1(z) = q_1 \end{aligned} \right\} \quad (4)$$

with the condition that $n_i(z)$ and $[dn_i(z)]/dz$ are continuous at $z = 0$ and that the solution is periodic in z .

Here

$$(f)_{ik} = [i! (i+1)! k! (k+1)!]^{-1/2} \int_0^\infty f(x) L_i^{(1)}(x) L_k^{(1)}(x) x e^{-x} dx;$$

$$q_k = \frac{3}{\lambda_s} [k! (k+1)!]^{-1/2} \int_0^\infty L_k^{(1)}(x) \frac{\Psi(x)}{V x} \left[\frac{1}{T(x)} - \frac{1}{T^*} \right] dx \approx \frac{3}{\lambda_s} \frac{\rho \sigma_0 \pi \Gamma}{2} \Psi(x_p) \frac{L_k^{(1)}(x_p)}{V k! (k+1)!},$$

where ρ is the density of resonant-absorbent nuclei, σ_0 is the resonance cross section ($x = x_p$), $\Gamma k \theta$ is the total width of the level, $T(x)$ is the lifetime of neutrons with energies $E = x k \theta$. The solution has the form

$$\left. \begin{aligned} n_0^\gamma(z) &= \left\{ \frac{1}{k_1 - k_2} \left\{ k_1 \left[C_2 \operatorname{sh} \alpha_2 z + D_2 \operatorname{sh} \alpha_2 z + \frac{F_2}{\alpha_2^2} (\operatorname{ch} \alpha_2 z - 1) \right] - \right. \right. \\ &\quad \left. \left. - k_2 \left[C_1 \operatorname{ch} \alpha_1 z + D_1 \operatorname{ch} \alpha_1 z + \frac{F_1}{\alpha_1^2} (\operatorname{ch} \alpha_1 z - 1) \right] \right\} \right\}_\gamma; \\ n_1^\gamma(z) &= \left\{ \frac{1}{k_1 - k_2} \left\{ \left[C_1 \operatorname{ch} \alpha_1 z + D_1 \operatorname{sh} \alpha_1 z + \frac{F_1}{\alpha_1^2} (\operatorname{ch} \alpha_1 z - 1) \right] - \right. \right. \\ &\quad \left. \left. - \left[C_2 \operatorname{ch} \alpha_2 z + D_2 \operatorname{sh} \alpha_2 z + \frac{F_2}{\alpha_2^2} (\operatorname{ch} \alpha_2 z - 1) \right] \right\} \right\}_\gamma. \end{aligned} \right\} \quad (5)$$

Here γ is the plate number;

$$\begin{aligned} \alpha_{i\gamma}^2 &= (a_{11} + k_i a_{12})_\gamma; \\ F_{i\gamma} &= (q_0 + k_i q_1)_\gamma; \\ a_{11}^\gamma &= \frac{3}{\lambda_s} \left[\frac{1}{V x T(x)} \right]_{00}^\gamma; \\ a_{12}^\gamma &= a_{21}^\gamma = \frac{3}{\lambda_s} \left(\frac{1}{V x T(x)} \right)_{10}^\gamma; \\ a_{22}^\gamma &= \frac{3}{\lambda_s} \left(\frac{1}{V x T(x)} \right)_{11}^\gamma + \frac{6}{M \lambda_s^2}, \end{aligned}$$

and k_i are the roots of the equation $k^2 + k \left(\frac{a_{11} - a_{22}}{a_{12}} \right) - 1 = 0$. In the numerical analysis the values of the

parameters were taken from [3] for graphite and Pu^{239} : $a = 2$, $b = 10$ cm, $\frac{3}{\lambda_s} \frac{\rho \sigma_0 \pi \Gamma}{2} \Psi(x_p) = 1$, the concentra-

tion of Pu^{239} nuclei was equal to 0.0044. The notion of series convergence gives a ratio $n_1(z)/n_0(z)$ which is a maximum for $z = a$. With $\theta = 300^\circ\text{K}$, $n_1(a)/n_0(a)$ does not exceed 10% and falls off rapidly with increasing temperature. From this fact it is clear that the series (2) will converge quite rapidly.

2. Let us now consider the distribution function for neutrons from a point source in a bounded volume containing a uniform mixture of retarder nuclei and nuclei that resonate in the super-thermal region. The solution $N(r, x)$ is found in the form of an expansion in the eigenfunctions $R_C(r)$ of the Laplacian of the corresponding boundary problem $[\Delta R_l(r) + \Omega_l R_l(r) = 0]$, i.e.,

$$N(r, x) = \sum_l R_l(r) \Phi_l(x). \quad (6)$$

For $\Phi_l(x)$ we obtain the equation

$$\begin{aligned} x \frac{d^2 \Phi_l(x)}{dx^2} + x \frac{d \Phi_l(x)}{dx} + \Phi_l(x) - \frac{M \lambda_s^2}{6} \Omega_l \Phi_l(x) - \\ - \frac{M \lambda_s}{2} \frac{\Phi_l(x)}{V x T(x)} + Q_l(x) = 0. \end{aligned} \quad (7)$$

where $\Psi_l(x)$ is the solution of the equation $\Phi_l(x) = \Psi_l(x) + \varphi_l(x)$, $\Phi_l(x)$ is written in the form

$$\begin{aligned} x \frac{d^2 \Psi_l(x)}{dx^2} + x \frac{d \Psi_l(x)}{dx} + \Psi_l(x) - \frac{M \lambda_s^2}{6} \Omega_l \Psi_l(x) - \\ - \frac{M \lambda_s}{2} \frac{\Psi_l(x)}{V x T^*} + Q_c(x) = 0. \end{aligned} \quad (7a)$$

Green's function for (7a) is given in [4] for the case of weak capture. The function $\varphi_l(x)$ is found in the form of the expansion (2), and for the coefficients of the expansion C_k we obtain the system of equations

$$\begin{aligned} C_h \left(k + \frac{M \lambda_s^2}{6} \Omega_l \right) + \frac{M \lambda_s}{2} \sum_{i=1}^{\infty} C_i \left(\frac{1}{V x T(x)} \right)_{ih} + \\ + \frac{L_h^{(1)}(x_p)}{V \kappa! (k+1)!} = 0 \quad (\text{at } k=0, 1, \dots). \end{aligned} \quad (8)$$

Here

$$\frac{M \lambda_s}{2} \frac{\rho \sigma_0 \pi \Gamma}{2} \Psi_0(x_p) = 1.$$

The numerical calculation was carried out in the two-group approximation for the fundamental harmonic of the distribution function in a spherical retarder for the same values of the parameters as in the first example and for two values of the radius: $R_1 = 62.8$ and $R_2 = 31.4$ cm. The results of the numerical calculation are shown in the accompanying table.

At a temperature of $\theta = 500^\circ\text{K}$ the calculation was also made in the three-group approximation: for $R = R_1$, $C_0 = 0.886$, $C_1 = 0.090$, $C_2 = 0.049$, and for $R = R_2$, $C_0 = 0.426$, $C_1 = 0.165$, $C_2 = 0.092$. From these results it follows that the convergence of (2) at not-too-low temperatures proves to be once again satisfactory.

Results of Numerical Analysis

Parameter	Retarder temperature, °K					
	300		500		900	
	R_1	R_2	R_1	R_2	R_1	R_2
C_0	-11,92	-4,05	-0,955	-0,543	-0,196	-0,170
C_1	2,76	3,12	0,118	0,221	0,008	0,025

In conclusion, the authors are indebted to M. V. Kazarnovskii for suggesting the topic and to Z. P. Mukhina for carrying out the numerical calculations.

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A RING CYCLOTRON ACCELERATOR WITH VERTICALLY GROWING MAGNETIC FIELD

A. P. Fateev and B. N. Yablokov

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Several attempts have been undertaken recently [1-3] to avoid using magnetic fields having a symmetry plane, in particle accelerators, and to change over to fields increasing in the vertical direction. In the latter case, the particle would be displaced upward as energy increases, the perimeter of the particle orbit remaining constant so that the restraint $\omega(E) = \text{const}$, where ω is the frequency of revolution of the particle, would hold even for relativistic energies. Keeping the frequency of revolution constant makes it possible to produce particle currents attainable only in cyclotrons at present. However, the authors consulted [1-3] have failed to demonstrate the possibility of stable particle acceleration under such conditions.

Below we consider an accelerator with a vertically growing guide magnetic field constant in time (a ring cyclotron), and investigate the stability of particle motion in this machine.

The magnetic system of the ring cyclotron consists of identical periodicity elements each of which is made up of two sectors spaced by rectilinear intervals which are field-free. The directions of the magnetic field in adjacent sectors are taken to be opposed, so that the curvature of the orbit changes sign. The fields in both sectors increase in absolute value along the vertical in obedience to the law $H \sim z^n$. Under certain conditions, alternation of the sign of the field may bring about particle focusing. At the same time, growth of the field along the vertical displaces the particle orbit upward as energy is increased, with the perimeter of the orbit remaining constant.

An accelerator of the type under discussion may be devised in various ways. We shall restrict our attention to the simplest case (Fig. 1), where the particle orbit describes a plane curve consisting of arcs of circles having curvatures of alternating sign, and of the straight-line segments $R_1 = R_2 = R = \text{const}$. If the angular dimensions of sectors ν_1 and ν_2 are large compared to the rectilinear intervals of the radial clearance, then fringe effects may be neglected.

Let the magnetic field in the region of such an orbit have the form (Fig. 2):

$$\left. \begin{aligned} H_z &\approx \pm H_0 \left[1 + ny + \frac{n^2}{2} (y^2 - x^2) \right]; \\ H_r &\approx \pm H_0 nx [1 + (n+1)y]; \\ H_\theta &\approx 0, \end{aligned} \right\} \quad (1)$$

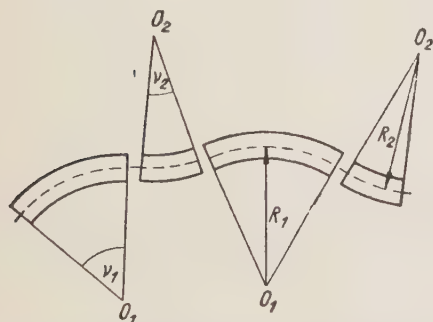


Fig. 1. Arrangement of sectors, and the particle orbit in a ring cyclotron.

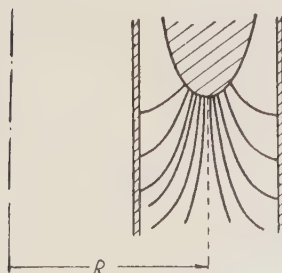


Fig. 2. Field configuration in the ring cyclotron.

where $x = \rho/R$, $y = \Delta z/R$, are the dimensionless deviations from the prescribed orbit, and $\underline{n}$ is a parameter determining the configuration of the magnetic field.* (Here and in what follows, the upper sign refers to sector ν_1 , and the lower sign refers to sector ν_2). Now, by linearizing the equation of motion in the field (1), we obtain

$$\left. \begin{aligned} \frac{d^2 x}{d\vartheta^2} + x \pm ny &= 0; \\ \frac{d^2 y}{d\vartheta^2} \pm nx &= 0 \end{aligned} \right\} \quad (2)$$

(the angle ϑ is measured from the center of curvature). The solution of this system of equations in both sectors may be written in the form

$$x = \sum_{i=1}^4 x_i e^{\omega_i \vartheta}; \quad y = \sum_{i=1}^4 y_i e^{\omega_i \vartheta}, \quad (3)$$

where $\omega_i = \pm \sqrt{-\frac{1}{2} \pm \sqrt{\frac{1}{4} + n^2}}$, with x_i and y_i constant and determined by the initial conditions, are interrelated by the equation

$$y_i = -\frac{n}{\omega_i} x_i. \quad (4)$$

By linking up solutions (3) at the boundaries of the sectors of each m -th periodicity element while assuming the rectilinear intervals to be absent, we obtain the matrix equation

$$x_{m+1} = M x_m; \quad x = \begin{pmatrix} x_1 \\ x_2 \\ x_3 \\ x_4 \end{pmatrix}, \quad (5)$$

whose solution is of the form

$$x_m = \Lambda^m x_0. \quad (6)$$

To determine Λ , we have the characteristic equation

$$\Lambda^4 - S_1 \Lambda^3 + S_2 \Lambda^2 - S_3 \Lambda + S_4 = 0, \quad (7)$$

where S_i is the sum of all possible diagonal minors of i -th order of the matrix M . According to Liouville's theorem, $S_4 = 1$ in the case under discussion. Further, it may be shown that the coefficients S_i (with $i = 1, 2, 3$) are real and that $S_1 = S_3$. Equation (7) is thus a fourth-order recursive equation with real coefficients. The solution of Eq. (6) will be a stable one, if all the Λ_i are complex. This leads to the condition $4S_2 - S_1^2 > 8$.

As an example, the stability region is determined for a ring cyclotron having the following parameters: $N = 30$, $n \approx 10$. In this case, there exists a sufficiently wide region of stability at $1.21 < \nu_1/\nu_2 < 1.33$. Depending on the choice of operating point, the gain of the machine will vary from ~ 7 to 10 . Note that the mean orbital radius is determined by the finite energy of the accelerator.

*The index $\underline{n}$ representing the growth of the magnetic field along the vertical is obviously dependent on the choice of the origin $\underline{z}$ of the frame of reference.

In examining the spatial orbits, we might hope to obtain (by analogy with the "symmetric" ring synchro-cyclotron [4, 5]) a "symmetric" ring cyclotron in which acceleration of electrons in opposite directions and head-on collisions are possible simultaneously.

Because of the radiation emitted by the circulating electrons, there exists in the ring cyclotron, for an even phase distribution of particles, a set of critical orbits determined by the relation

$$\gamma \simeq 35 (eV_0 \cos \varphi_0 R)^{1/4}; \quad \gamma = \frac{E}{E_0}, \quad (8)$$

where the energy increment per revolution $eV_0 \cos \varphi_0$ is measured in electron volts, and the boundary values φ_0 are determined by the tolerances of the magnetic field. By bunching the electrons around one particular phase (e.g., utilizing a preliminary phase bunching operation or introducing the phase dependence of the frequency of revolution), it is possible to effect a significant reduction in the energy spread of the accelerated beam, and to stack up the particles in a practical manner.

The authors express their gratitude to A. A. Kolomenskii for his kind discussion of the experimental work.

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SOME PROPERTIES OF ACCELERATOR ORBITS WHERE SIMILITUDE IS OBSERVED

A. A. Kolomenskii and A. N. Lebedev

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The condition of dynamic similitude is the term used to describe the requirement that the characteristic numbers of betatron oscillations be independent of the energy of the particles being accelerated. The need for dynamic similitude is dictated by the fact that the frequencies of the betatron oscillations must not take on the dangerous resonance values during an acceleration cycle. It is especially important that the similitude condition be satisfied in fixed-field alternating-gradient accelerators, where the orbit parameters undergo considerable variations during acceleration.

As indicated in an earlier paper, the condition of dynamic similitude for plane orbits is the special configuration of the magnetic field

$$H_z = f(\theta) r^{n_0}, \quad (1)$$

where H_z is the component of the field at right angles to the orbit plane; $f(\theta)$ is an arbitrary periodic function; $\underline{r}$ and θ are the cylindrical coordinates of the point on the orbit; $n_0 = \text{const}$. In this note, we consider some of the general properties of the motion of particles in such systems, on which research and development work has been much intensified of late, in connection with the problem of particle storage and beam collisions.

Let there correspond to some particle momentum $\underline{p}$ a plane closed orbit $\underline{r}(\sigma)$ (where σ is a length of arc). The linearized equations for deviations along the normal (ρ) and along the binormal (z) from this orbit exhibit, as we know, the familiar form

$$\left. \begin{aligned} \varrho'' + K^2 R^2 (1 - n) \varrho &= 0; \\ z'' + K^2 R^2 n z &= 0, \end{aligned} \right\} \quad (2)$$

where the field index $\underline{n}$ is measured along the normal to the orbit; K is the orbit curvature, and differentiation is performed with respect to a generalized azimuth $\vartheta = \sigma/R$ (R being the mean radius of the orbit). The radius vector of another orbit, the latter in this case corresponding to a momentum $\underline{p} + \Delta \underline{p}$, may be represented as

$$\underline{r}_1(\sigma) = \underline{r}(\sigma) - \psi(\vartheta) \underline{n} \frac{\Delta p}{p}, \quad (3)$$

where $\underline{n}$ is a unit vector normal to the orbit; $\psi(\vartheta)$ is a periodic solution of the equation

$$\psi'' + K^2 R^2 (1 - n) \psi = K R^2 \quad (4)$$

(differentiation with respect to ϑ).

The invariance condition for equations (2) is obviously

$$\left(\frac{\partial KR}{\partial p}\right)_{\vartheta=\text{const}}=0; \quad \left(\frac{\partial n}{\partial p}\right)_{\vartheta=\text{const}}=0. \quad (5)$$

Taking in account the fact that the length of an arc of the curve $\mathbf{r}_1(\sigma)$ is

$$\sigma_1 = \sigma + \frac{\Delta p}{p} \int_0^\sigma K \psi d\sigma, \quad (6)$$

differentiating Eq. (3), and using the Serret-Frenet formulas, the first condition in Eq. (5) may be readily reduced to the form

$$K(\alpha - 1) - nK^2\psi - K' \left[\int_0^\vartheta K \psi d\vartheta - \alpha \vartheta \right] = 0, \quad (7)$$

where $\alpha = \frac{1}{2\pi} \int_0^{2\pi} K \psi d\vartheta$ is the factor of instantaneous orbits. In this formula, $\underline{n}$ may be conveniently expressed

in terms of n_0 with the aid of the identity

$$\begin{aligned} n_0 &= -\frac{r}{H_z} \frac{\partial H_z}{\partial r} = -\frac{r}{H_z} \left(\frac{\partial H_z}{\partial \varrho} \frac{d\varrho}{dr} + \frac{\partial H_z}{\partial \sigma} \frac{d\sigma}{dr} \right) = \\ &= -K(rn)n - \frac{K'}{KR}(r\tau) \end{aligned} \quad (8)$$

(where τ is the tangent vector).

For this orbit configuration, Eqs. (4) and (7) constitute an inhomogeneous system with periodic coefficients, and linear in unknowns ψn_0 and ψ . As we see from the general theorems in [2], such a system may have only one periodic solution in the nonresonant case, a solution having the same period as the coefficients. Direct verification readily convinces the reader that this solution is

$$n_0 = \text{const}; \quad \psi = -\frac{1}{1-n_0}(rn) = -\alpha(rn), \quad (9)$$

in which case the solution is not bound to any concrete shape of orbit, i.e., it remains valid for an arbitrary azimuthal dependence of the magnetic field.

The condition of dynamic similitude makes itself felt not only on the orbit geometry, but also exerts a material effect on the dynamics of the particles undergoing acceleration. In a number of cases, similar orbits reveal an analogy with circular orbits. This finds particularly clear reflection in the problem concerning radiation effects in fixed-field accelerators.

As has been shown in several previous communications on the subject, the radiation reaction leads to positive or negative damping of the betatron and synchrotron oscillations with a damping constant [3]:

$$\left. \begin{aligned} \xi_r &= \frac{\Gamma}{2} [1 - (1 - 2n)K\psi] \text{ (radial oscillations)} \\ \xi_z &= \frac{\Gamma}{2} \text{ (vertical oscillations)} \\ \xi_s &= \frac{\Gamma}{2} [2 + (1 - 2n)K\psi] \text{ (synchrotron oscillations)} \end{aligned} \right\} \quad (10)$$

where a bar over a symbol denotes averaging over a periodicity element, and Γ is the dimensionless intensity of radiation. The sum of the damping constants is always equal to $2\bar{\Gamma}$, independently of the form of the magnetic system.

If the orbit consists of arcs of constant curvature, as in conventional weak-focusing and strong-focusing synchrotrons, then we obtain the familiar formulas

$$\xi_r = \frac{\alpha}{2} \bar{\Gamma}; \quad \xi_z = \frac{\bar{\Gamma}}{2}; \quad \xi_x = \frac{4-\alpha}{2} \bar{\Gamma}, \quad (11)$$

from which, in particular, we learn that, in a strong-focusing magnet ($\alpha \ll 1$), radial betatron oscillations become unstable [3]. Applied to accelerators with similar orbits of alternating curvature, this conclusion is not quite so obvious, but retains its validity, as demonstrated below. The basic importance of this assertion must be stressed, since it signifies that the damping systems developed for conventional synchrotrons cannot be made effective under the similitude conditions [3, 4]. The proof reduces to substitution of Eq. (9) for the function ψ in Eq. (10). We then have:

$$\xi_r = \frac{\Gamma}{2} \left[1 + K\alpha(r\tau) - 2\alpha \left(n_0 + \frac{K'}{RK}(r\tau) \right) \right]. \quad (12)$$

Bearing in mind that $\Gamma \sim K^2$, we have

$$2 \frac{\Gamma K'}{RK}(r\tau) = - \frac{\Gamma}{R}(r\tau)' = -\bar{\Gamma} - \overline{K\Gamma(r\tau)}. \quad (13)$$

Substitution of Eq. (13) into (12) yields

$$\xi_r = \frac{\Gamma}{2} [1 + 2\alpha n_0 - \alpha] = \frac{\Gamma}{2} (\alpha - 1) = \frac{n_0}{1 - n_0} \frac{\Gamma}{2}. \quad (14)$$

The expressions for the remaining damping constants also agree with Eq. (11). It is important to note that these expressions are in no way connected to the azimuthal dependence of the magnetic field, which may be arbitrary, and are derived solely from the similitude condition.

Now, when similitude is preserved it is impossible to set up a damping system operating by the coupling between radial oscillations and synchrotron oscillations, so that a possible technique for suppressing radiation instability is still available in the artificial coupling of those oscillations with the vertical oscillations. Using the formulas derived in a previous paper [3], we find the damping constants of the bound oscillations in a similar system:

$$\xi_{r,z} = \frac{\bar{\Gamma}}{4} \left\{ \alpha \pm (2 - \alpha) \frac{\delta}{[\delta^2 + \Delta^2]^{1/2}} \right\}, \quad (15)$$

where δ is the distance of the operating point from the difference resonance of the coupling; Δ is a parameter characterizing the coupling between the two modes of oscillation. Numerical estimates show that, in order to bring about efficient damping in an electron ring synchrocyclotron of the type described in [5], a magnetic field comprising $\sim 10\%$ of the guide field must be introduced.

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MEASUREMENT OF THE RADIATIVE-CAPTURE γ -EMISSION SPECTRA OF NEUTRONS IN SOME ROCKS

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Gamma emission with a spectrum consisting of γ -lines characteristic of each nuclear species is the result when the nuclei of chemical elements are bombarded by thermal neutrons [1, 2]. Analysis of the spectra of γ -lines accompanying neutron capture sometimes allows us to infer the presence of certain elements in substances of complex chemical makeup [3]. In 1956, the authors completed experiments, in a bore hole, confirming the possibility of utilizing this method for detecting certain chemical elements in various rock species.

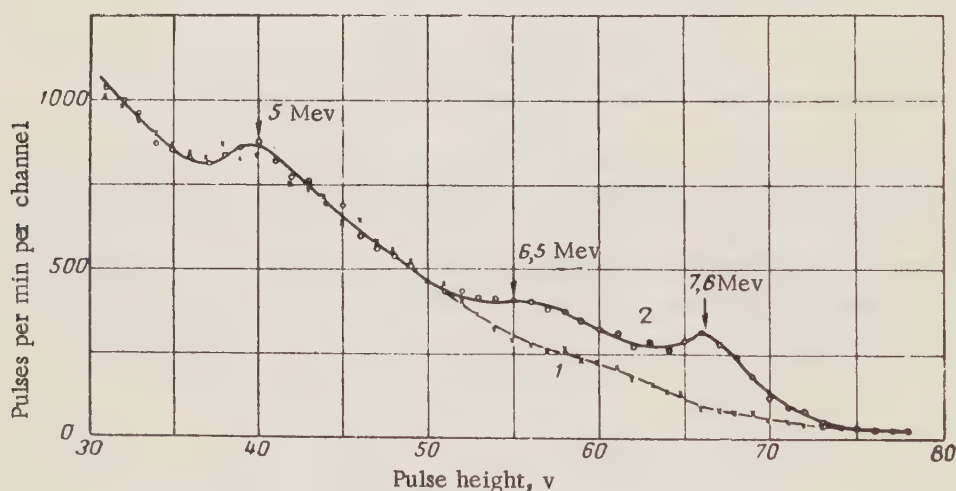


Fig. 1. γ -Emission spectra, for γ -radiation in response to neutron irradiation of cherts (1) and diorites (2).

The neutron source was a Po +Be preparation with a yield of $2 \cdot 10^6$ neutrons/sec. Gamma emission was recorded by means of a scintillation spectrometer. The spectrometer resolution for the Cs^{137} γ -line (0.66 Mev) was 12%. A NaI crystal, FEU-19M phototube multiplier, and an amplifier were placed in a logging tool such as is used in borehole logging. To minimize γ -radiation background due to neutron capture in the material of the logging tool the portion of the latter in which the neutron source, crystal, and phototube multiplier were housed was made of textolite. The crystal and photomultiplier were specially shielded by a layer of boron carbide. A bismuth screen was inserted between the source and crystal (spacing of 10 cm). The source was surrounded by paraffin to improve conditions favorable to slowing down the neutrons.

The γ -emission spectra were measured in the energy region from 4.5 Mev and higher, since the spectrum is distorted at lower energies on account of the scattering of the γ -radiation of the neutron source proper by the

surrounding rocks and drilling mud. Fig. 1 shows the pulse-height distribution of pulses corresponding to the γ -emission spectra, as a result of neutron irradiation of two rock species differing in composition: cherts, which consist basically of oxygen and silicon (curve 1), and diorites, which consist for the most part of oxygen, silicon, sodium, calcium, aluminum, and iron (curve 2). Transverse cross sections of thermal-neutron capture (σ) and the energy of the principal γ -lines, ($E_\gamma > 4.5$ Mev) for the elements mentioned are given in tabular form.

	O	Na	Al	Si	Ca	Fe
σ , barns	$2 \cdot 10^{-4}$	0,50	0,22	0,13	0,43	2,43
E_γ , Mev	—	6,41	7,70	4,95	6,42	7,64

Since oxygen absorbs virtually no neutrons, the γ -emission spectrum of the cherts in the energy region around 5 Mev shows a peak corresponding to the silicon γ -line (4.95 Mev).* The γ -emission spectrum of the diorites is observed to peak at some other points, corresponding to the γ -lines of sodium and calcium (6.4 Mev), aluminum and iron (7.6 Mev).

As we see from Fig. 1, the largest difference discernible in the γ -emission spectra of the cherts and the diorites is observed at energy ~ 7.6 Mev, corresponding to the γ -lines of aluminum and iron. By continuously recording γ -emission intensity from 7.6 Mev energy on, these elements can be readily identified in a bore-hole section.

In Fig. 2 we have the logging charts obtained from the same sample. Curve 2 corresponds to an integral count at a discrimination level of 0.1 Mev and repeats the logging density trace obtained with the Co^{60} source (curve 1) on which we discern cherts of low density. On the diagram corresponding to the integral count at discrimination level 4.5 Mev (curve 3), the cherts are conspicuous with their shallow minimum due to the absence of aluminum and iron in that species. A particularly sharp difference in the makeup of elements in cherts and diorites stands out in the diagram corresponding to a differential count over the energy range 7.3-9 Mev (curve 4).

The results so obtained provide confirmation of the possibility of determinations of individual chemical elements in rocks by the γ -emission resulting from radiative capture of neutrons.

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*This is due to the high silicon content in cherts, which acts to offset the relatively small capture cross section.

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SLOWING DOWN OF NEUTRONS IN STEEL-WATER MIXTURES

L. A. Geraseva and V. V. Vavilov

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Measurements of the second spatial moment of the slowing-down density of fission neutrons in water and steel-water mixtures was carried out in a steel tank measuring 74 cm by 74 cm by 100 cm, flooded with water and containing slabs of St-3 steel 71.5 cm by 71.5 cm by 0.3 cm.

To forestall corrosive attack on the slabs and tank walls, these components were finished with a bakelite resin. The slabs were placed in the tank at right angles to the direction of measurement of the slowing-down density distribution, and were fastened in the required position by means of duralumin and plexiglas racks placed on the bottom and walls of the tanks.

Measurements were performed for three concentrations by specific volume of iron and water ($\rho = \frac{\text{iron volume}}{\text{iron volume plus water volume}}$), equal to 0.14, 0.26, and 0.43. A control experiment was staged to measure the neutron age in water. The fission-neutron source used was a converter which converted thermal neutrons from the pile into neutrons of the U^{235} fission spectrum, and was made of uranyl uranate 75% enriched with U^{235} . The spatial distribution of slowing-down neutrons was measured by means of cadmium-plated indium foils (mean thickness 40 mg/cm²).

The relatively weak flux of thermal neutrons, and consequently of fast neutrons as well, emerging from the converter, was not sufficient to carry out measurements at distances greater than 56 cm from the source, as required to determine the age of the neutrons. It is a known fact that the slowing-down density at large distances from the source falls off in obedience to the law $\sim (ke^{-r/\lambda})/r^2$, where λ is the relaxation length. This circumstance was used to advantage in extrapolating the distributions to infinity.

To compute the neutron age τ , we used the familiar formula

$$\tau = \frac{1}{6} \frac{\int_0^\infty Ar^4 dr}{\int_0^\infty Ar^2 dr}.$$

The values of $\int_0^R Ar^4 dr$ and $\int_0^R Ar^2 dr$ were determined empirically, and the values of $\int_R^\infty Ar^4 dr$ and $\int_R^\infty Ar^2 dr$ were obtained analytically by extrapolating in accord with the law $A \sim (ke^{-r/\lambda})/r^2$. The value of

k was arrived at by choosing a function $ke^{-r/\lambda}$ such that the extrapolated portion could be "tacked on" to the experimental portion; the value given to λ was taken from the last points of the experimental distribution.

The values obtained were:

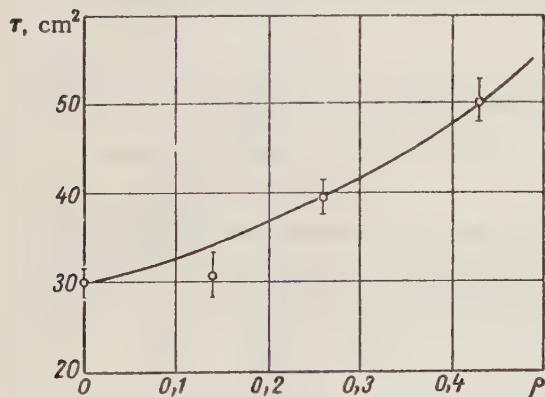
$$\begin{aligned}\tau_{\text{H}_2\text{O}} &= 30,2 \pm 1,5 \text{ cm}^2 \\ \tau_{\text{Fe}+\text{H}_2\text{O}} &= 31,0 \pm 2,7 \text{ cm}^2 \quad (\varrho=0,14); \\ \tau_{\text{Fe}+\text{H}_2\text{O}} &= 39,7 \pm 2,0 \text{ cm}^2 \quad (\varrho=0,26); \\ \tau_{\text{Fe}+\text{H}_2\text{O}} &= 50,4 \pm 2,5 \text{ cm}^2 \quad (\varrho=0,43).\end{aligned}$$

The neutron age in a mixture of several substances, each of which has a known slowing-down length is calculated from the formula

$$\frac{1}{\tau} = \sum_i \frac{\varrho_i^2}{\tau_i} + \sum_{i>j} \frac{\varrho_i \varrho_j}{\sqrt{\tau_i \tau_j}} \times \left(\sqrt{\frac{1 - \frac{2}{3A_{i'}}}{1 - \frac{2}{3A_i}}} \frac{\tau_i}{\tau_{i'}} + \sqrt{\frac{1 - \frac{2}{3A_i}}{1 - \frac{2}{3A_{i'}}}} \frac{\tau_{i'}}{\tau_i} \right), \quad (1)$$

where ρ_i is the specific concentration by volume of the i -th substance present in the mixture.

On the diagram, the curve corresponds to predicted values of τ up to energy 1.46 eV for various concentrations of the mixture of iron and water laminations, computed by Eq. (1). In the calculations, the following values of τ were used: for iron $\tau = 743 \text{ cm}^2$ (calculated) for water $\tau = 30,5 \text{ cm}^2$ (empirical). As we see from the accompanying diagram, the values arrived at empirically for the age of neutrons in steel-water laminations show good fit with predicted values.



Dependence of neutron age in steel-water laminations on concentration. —) plotted by

computation; $\bigcirc$) empirically derived data points.

In conclusion, the authors would like to avail themselves of this opportunity to express their acknowledgment to B. G. Dubovskii, Yu. A. Sergeev for formulating the problem and for their kind participation in the discussion of the results, and to our co-workers V. K. Labuzov, Yu. S. Ziryukin, M. M. Kuzichkina, A. T. Anfilatov, who took part in the measurements.

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DETERMINATION OF DEGREES OF EQUILIBRIUM OF SHORT-LIVED RADON DAUGHTERS IN AIR

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It was demonstrated earlier [1-3] that the overwhelming portion of the absorbed dose in the case of radon inhalation is due to the short-lived daughter products of radon decay.

The degree of equilibrium of the short-lived daughter products of radon decay* was determined by the usual method of filtering the air with subsequent measurement of the filters for α -radiation [4]. The amount of radon present was determined by sampling the air in an ionization chamber (known in this case as an emanation chamber) and measuring the amount of ionization current.

Below, we suggest a method for determining the concentration of short-lived daughter products of radon decay which is based solely on measurements of the amount of ionization current flowing in the emanation chamber.

We shall deal only with the ionization current due to α -emitting isotopes (RaA and RaC'), on the basis that the energy of the α - particles in such a case is much higher than the energy of the β -radiation.

We use the term q to denote the radon concentration, in curies/liter, and η_A , η_B , and η_C to denote the degrees of equilibrium of RaA, RaB, and RaC, respectively.

The activity due to RaA alone, i.e., the RaA present in a unit volume of the chamber, is $q\eta_A e^{-\lambda_A t}$ (λ_A , λ_B , and λ_C being the decay constants for RaA, RaB, and RaC, respectively), and the RaA activity for RaA formed from radon in a unit volume of the chamber is $-q(1 - e^{-\lambda_A t})$. The total activity due to RaA is

$$A_A = qv [\eta_A e^{-\lambda_A t} + 1 - e^{-\lambda_A t}], \quad (1)$$

where v is the chamber volume.

The activity due to RaC' in the chamber at any instant is $A_{C'}(t) = A_C(t)$, and will be a sum of the activities due to RaC (A_C^I), RaB (A_C^{II}), RaA (A_C^{III}), and Rn (A_C^{IV}). Using the solution for a chain of radioactive elements [5], we obtain

$$A_C^I(t) = q\eta_C e^{-\lambda_C t}; \quad (2)$$

$$A_C^{II}(t) = q\eta_B \frac{\lambda_C}{\lambda_C - \lambda_B} (e^{-\lambda_B t} - e^{-\lambda_C t}); \quad (3)$$

*The "degree of equilibrium" of a given radon daughter refers to the ratio of the amount of daughter product to the equilibrium ratio in air (translator's note).

$$A_C^{III}(t) = q\eta_A\lambda_B\lambda_C \left[\frac{e^{-\lambda_A t}}{(\lambda_B - \lambda_A)(\lambda_C - \lambda_A)} + \frac{e^{-\lambda_B t}}{(\lambda_A - \lambda_B)(\lambda_C - \lambda_B)} + \frac{e^{-\lambda_C t}}{(\lambda_A - \lambda_C)(\lambda_B - \lambda_C)} \right]; \quad (4)$$

$$A_C^{IV}(t) = q \left[1 - \frac{\lambda_B\lambda_C e^{-\lambda_A t}}{(\lambda_B - \lambda_A)(\lambda_C - \lambda_A)} - \frac{\lambda_A\lambda_C e^{-\lambda_B t}}{(\lambda_A - \lambda_B)(\lambda_C - \lambda_B)} - \frac{\lambda_A\lambda_B e^{-\lambda_C t}}{(\lambda_A - \lambda_C)(\lambda_B - \lambda_C)} \right]. \quad (5)$$

Turning now to the amount of ionization current due to the entire volume of the chamber,

$$I = k \left(\frac{E_A}{E_{Rn}} I_A + \frac{E_{C'}}{E_{Rn}} I_{C'} \right) v \quad (6)$$

we must bear in mind that the amounts of energies of α -particles from RaA and RaC' are different from E_{Rn} , in consequence of which the ratios E_A/E_{Rn} and $E_{C'}/E_{Rn}$ appear in the expressions for the ionization current (and k in the above formula is a conversion factor from activity to current). Finally, we have

$$I = kqv [f_{Rn}(t) + \eta_A f_A(t) + \eta_B f_B(t) + \eta_C f_C(t)] = kqv F(t). \quad (7)$$

The graphs plotted for functions $f_{Rn}(t)$, $f_A(t)$, and $f_C(t)$

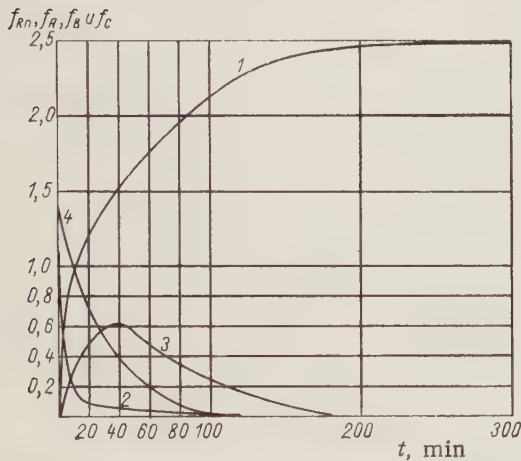


Fig. 1. Graphs of functions $f_{Rn}(t)$, $f_A(t)$, $f_B(t)$, and $f_C(t)$ (curves 1-4, respectively).

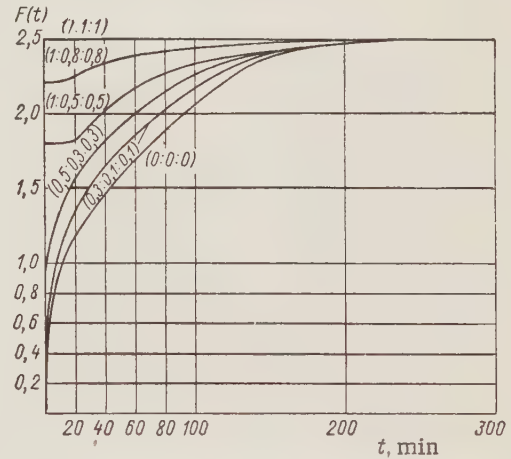


Fig. 2. Function $F(t)$ at different values of η_A , η_B , η_C . Parentheses enclose the ratios of the degrees of equilibrium, as $\eta_A : \eta_B : \eta_C$.

give some idea of the contributions of radon, RaA, RaB, and RaC, respectively, to the amount of ionization current, and may be seen in Fig. 1.

The ratios η_A , η_B , and η_C may be found from Eq. (7), if three values of the ionization current taken at different instants are used. The values of $f_{Rn}(t)$, $f_A(t)$, $f_B(t)$, and $f_C(t)$ for any instant of time t are read off from the graphs in Fig. 1.

Fig. 2 shows a graph of the function $F(t)$, which gives us some idea of the nature of the increase in ionization current flowing in the chamber at different values of η_A , η_B , and η_C . As we see from Fig. 2, the ionization current values in the chamber corresponding to different ratios of the degrees of equilibrium η_A , η_B , and η_C differ markedly from each other in the first 60-80 min following sampling of the air.

The low curve corresponds to the increase in current attributable to radon without decay products. In the case where the decay products are in equilibrium with radon ($\eta_A = \eta_B = \eta_C = 1$), $F(t) = \text{const.}$

The method described above for determining the concentration of short-lived daughter products of radon decay is quite simple and requires no equipment other than conventional electrometers used for emanations (SG-IM). This method is suitable for measuring η_A , η_B , and η_C in rooms of fairly small volume.

A similar method may be employed for determining the concentrations of daughter products of other emanations.

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*Original Russian pagination. See C. B. translation.

LUMINESCENT DOSIMETERS BASED ON THE $\text{CaSO}_4 \cdot \text{Mn}$ PHOSPHOR FOR THE DETECTION OF GAMMAS, BETAS, AND NEUTRONS

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In 1951, Watanabe [1], in an investigation of the thermoluminescent behavior of $\text{CaSO}_4 \cdot \text{Mn}$, noted the ability of this phosphor to store up energy when subjected to γ -emission from radium. The possibility of using $\text{CaSO}_4 \cdot \text{Mn}$ in the dosimetry of ionizing radiations was discussed in several subsequent papers [2-4], but detailed quantitative data on the dosimetric properties of this phosphor are not available from any source. The present contribution seeks to fill that gap.

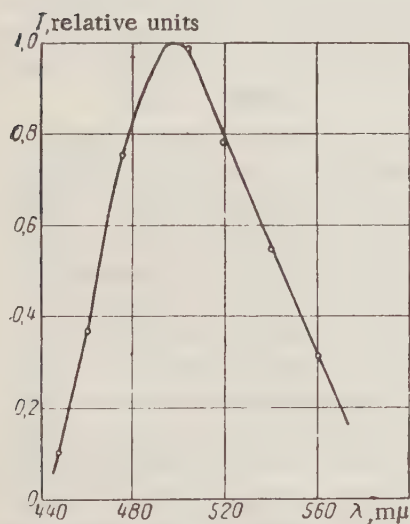


Fig. 1. Thermoluminescence spectrum of the phosphor $\text{CaSO}_4 \cdot \text{Mn}$.

The energy stored by the $\text{CaSO}_4 \cdot \text{Mn}$ phosphor during the irradiation process (and known as the light sum) may be obtained in the form of visible radiation in response to heating of the phosphor. The peak in the thermoluminescence spectrum of the phosphor falls in the region of 500 mμ (Fig. 1). The dependence of the degree of glow intensity on the temperature to which the excited phosphor is heated is shown in Fig. 2. The lone peak on the thermoluminescent glow curve in the temperature region higher than room temperature is independent of the mode of excitation. It might thus be inferred that the thermal properties of dosimeters using the $\text{CaSO}_4 \cdot \text{Mn}$ phosphor will be identical for different modes of excitation.

The response of $\text{CaSO}_4 \cdot \text{Mn}$ to x-radiation and soft γ -radiation is appreciably higher than its response to harder γ -radiation (curve 1 in Fig. 3). Using a lead filter of predetermined thickness, the dosimeter response was made to equal the response of the phosphor over a rather broad range (0.1-2.6 Mev) of energies (curve 2 in Fig. 3). The response of the phosphor to γ -radiation in the energy region of interest is so high that dosimeters having a luminescent surface area of 2 cm² are capable of measuring doses starting as low as 0.001 r, with unsophisticated photoelectric equipment to aid in the measurements. The upper range of measurable doses D is bounded by a break

in the linearity of the relation between the value of the light sum L and the irradiation dose. As we learn from Fig. 4a, this boundary lies in the region of irradiation doses of the order of several hundred roentgens; the sublinearity of the L(D) excitation curve stays within 30% even at $D \approx 1000$ r, a fact which may be taken into account in the measurements.

By utilizing the same sensing equipment as in γ -dosimetry, and the same luminescent surface area of the dosimeters, we were able to record doses of β -radiation (from $\text{Sr}^{90}\text{Y}^{90}$) ranging from $1 \cdot 10^5$ to $1 \cdot 10^8$ particles/cm². No break in the linearity of the L(D) relationship was observed, practically speaking (Fig. 4b).

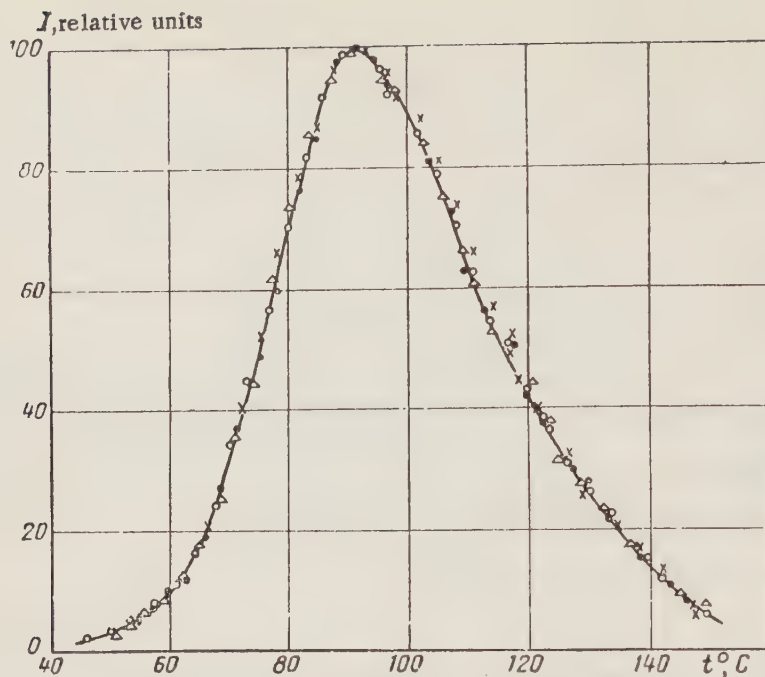


Fig. 2. Thermoluminescent glow curve of the phosphor $\text{CaSO}_4\cdot\text{Mn}$ in response to β -particles ($\bullet$), UV radiation ($\times$), x-rays (Δ), and gamma radiation ($\circ$).

According to preliminary data, the response of $\text{CaSO}_4\cdot\text{Mn}$ in the region of x-radiation from energy ~ 15 kev (source: an x-ray tube, 15 kev, copper anticathode) makes it possible to measure irradiation doses at the microroentgen level. The $L(D)$ plot for this case is seen in Fig. 4c. The plot of $\text{CaSO}_4\cdot\text{Mn}$ response vs hardness of radiation in the region of soft x-radiation and intermediate-range radiation, as well as the possibility of compensating the variation with hardness with the aid of filters, will be the subject of a subsequent paper.

Despite the fact that the peak on the thermoluminescence glow curve of the phosphor is found at 90-100°C, prolonged shelf storage of the irradiated phosphor even at room temperature results in partial loss of the light sum stored by the phosphor (decay curve, curve 1 in Fig. 5). In the case of the $\text{CaSO}_4\cdot\text{Mn}$ phosphor, the degradation of the light sum with time depends

neither on the magnitude nor on the dose rate of the irradiation received, in contrast to the light sum of the $\text{SrSeu}\cdot\text{Sm}$ phosphor, also used in individual dosimetry [5]. With increase in temperature, the decay in light sum with time is speeded up (curves 2 and 3 in Fig. 5). If the operating temperature of the dosimeter does not exceed 25°C and readings are taken daily, light-sum losses do not exceed 25% for a working day. However, with the increase in temperature and longer storage time for the dosimeter, losses may reach 50% and higher. As investigations have shown, the initial rapid decay $L(t)$ proceeds via luminescent emission of the light sum stored up at shallow trapping levels of the $\text{CaSO}_4\cdot\text{Mn}$ phosphor. By using a part of the light sum accumulated at deeper electron trapping levels for dose measurements, it would obviously be possible to improve the thermal characteristic of the dosimeter. The light sum to be measured then consists, as shown by calculations, of not less than 20% of the total (if the excitation temperature and the tem-

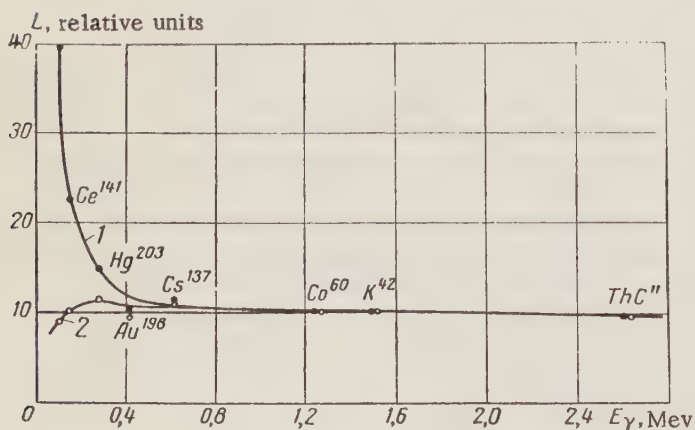


Fig. 3. Response of $\text{CaSO}_4\cdot\text{Mn}$ phosphor (1) and dosimeter (2) to γ -emission over an energy range.

perature and the tem-

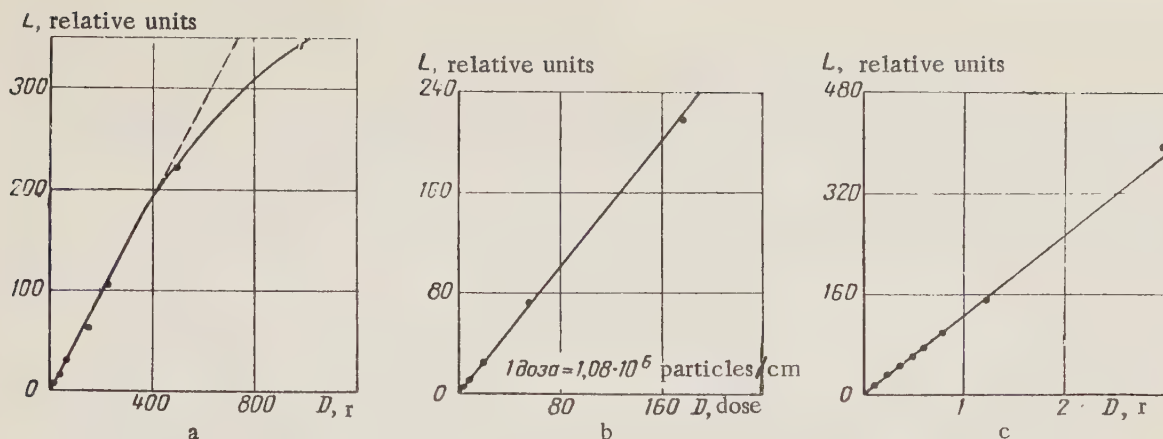


Fig. 4. Dependence of the magnitude of the stored light sum L on the dose of ionizing radiation for the cases of excitation by γ -radiation (a), x-radiation (b), and beta particles (c).

perature at which the dosimeters are left unexposed is not above 35°C) and may be retained for several days in the dosimeter. In practice, the thermally unstable portion of the light sum may be removed by pre-heating the dosimeter. This heating may constitute part of the over-all process of heating the dosimeter in measuring dosage.

The technology involved in fabrication of the phosphor is quite simple and yields readily reproducible results. The original materials are inexpensive and require no special purification, since impurities of heavy metals have no effect, even in relatively high concentration, on the response of the phosphor to ionizing radiations.

Repeated bombardments by radiation and repeated heating of the phosphor fail to produce any appreciable alteration of its properties. A dosimeter based on the $\text{CaSO}_4\cdot\text{Mn}$ phosphor may function without being recalibrated for several years.

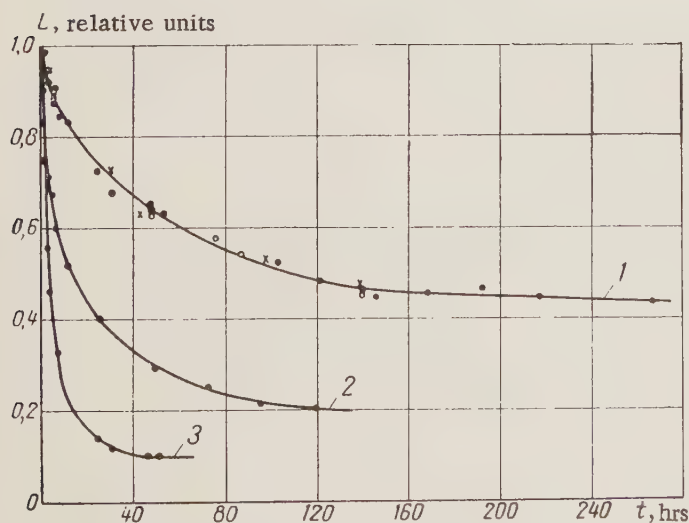


Fig. 5. Decay curve of light sum L with dosimeter left unexposed, as a function of temperature, amount and dose rate of irradiation: 1) 22°C; 2) 37°C; 3) 57°C; 20-r dose, dose rate in r/hr; (●) 6.7; (x) 1040; (○) 154.

In contrast to the SrSeu.Sm used in individual luminescent dosimetry [5], the $\text{CaSO}_4\cdot\text{Mn}$ phosphor is stable to moisture attack, and does not require any special leakproof container. Total lack of response to visible and ultraviolet radiation right up to wavelength $\lambda = 1500 \text{ \AA}$ may be counted as one of the additional advantages of this type of luminescent dosimeter. High-density emission in the region 2600-1800 $\text{\AA}$ (not less than 1 mw/cm^2) with prolonged irradiation of the phosphor leads to partial radiationless loss in light sum. However, neither direct radiation from the sun at the level prevailing at the earth's surface nor, a fortiori, the light from an incandescent lamp, have any effect on the light sum accumulated.

The possibility of growing luminescent $\text{CaSO}_4\cdot\text{Mn}$ single crystals of modest size has been reported [4] in the literature. Such single crystals without supplementary crystal holders may find application in beta dosimetry. The use of single crystals in gamma dosimetry has

obviously made it possible to improve the sensitivity of the method by increasing the thickness of the layer of phosphor transparent to luminescent emission.*

The $\text{CaSO}_4\text{.Mn}$ phosphor may also be employed to record thermal and fast neutrons. In the first case, the lead filter is replaced by a filter made of a thin cadmium wafer. To record fast neutrons, polymethyl methacrylate is introduced into the composition of the phosphor after the latter has been fabricated.

The totality of all the properties of the $\text{CaSO}_4\text{.Mn}$ phosphor discussed above justify us in viewing it as one of the most promising developments in individual luminescent dosimetry.

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*Measurements have shown that, for the usual grain size of $\text{CaSO}_4\text{.Mn}$ powder ranging from 1-5 μ , an increase of over 0.5 mm in the thickness of the layer of powder would not be effective.

SCIENCE AND ENGINEERING NEWS

LETTER FROM A READER

L. A. Artsimovich

(On the article "Entropy Trapping of a Plasma by a Reversal of the Magnetic Bottle Configuration")

Translated from Atomnaya Énergiya, Vol. 8, No. 6, p. 562,

June, 1960

Original article submitted April 15, 1960

In the third issue of Atomnaya Énergiya for the present year, there appeared an abstract in the section "Science and Engineering News" under the signature of G. B., dealing with several articles by J. Tuck. It is stated therein that "J. Tuck . . . proposed a new method for injecting a plasma into a magnetic trap with an inversion of the magnetic bottle configuration . . ." In writing this, the author of the abstract was in error, since a technique of injection of plasma globs into magnetic traps having the described field configuration was studied much earlier by S. Yu. Luk'yanov and I. M. Podgornyi [1].

In this connection, I should like to note that this method had already been developed experimentally by our group at the Institute of Atomic Energy during the past few years. Eight months ago, even before the arrival of J. Tuck's first published article on the subject, the article by S. Yu. Luk'yanov and I. M. Podgornyi dealt with the problem of the possibility of trapping a plasma in a trap having a magnetic field of the configuration referred to, which the author of the abstract terms an anti-bottle configuration. The fact that theoretical research and experimental investigations have been underway in the Soviet Union along these trends has received mention earlier yet, in particularly in the papers presented by Soviet physicists at the second Geneva conference on the peaceful uses of atomic energy [2, 3].

It is difficult for me to judge whether or not J. Tuck was aware of the papers of the Soviet reseachers in this field, and in particular of this paper by S. Yu. Luk'yanov and I. M. Podgornyi, of which J. Tuck's work is a further development.

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GENERATION OF A Σ^- -HYPERON BY NEGATIVE PIONS WITH A MOMENTUM OF 8.3 Bev/c

M. I. Solov'ev

Translated from *Atomnaya Énergiya*, Vol. 8, No. 6, pp. 562-563,
June, 1960

A team of physicists working at the High Energies Laboratory of the Joint Institute for Nuclear Research (Dubna), integrated by V. I. Veksler, N. M. Viryasov, E. N. Kladnitska, A. A. Kuznetsov, A. V. Nikitin, M. I. Solov'ev (USSR), Wang Hang-Chang, Wang Tsu-Ren, Ting Ta-Tsao (China), Nguyen Dien Thu (Vietnam), A. Michula (Rumania), Kim Hi Un (Korea), Jiri Vran (Czechoslovakia), are conducting research with beams of negative π -mesons in a 24-liter propane bubble chamber. The bubble chamber is mounted inside a magnet configuration, the magnet having a constant field of 13,700 strength. The chamber has been in operation for over a year; during that time a large number of photographs have been amassed.

In an analysis of 40,000 photographs obtained in the beam of negative π -mesons with momentum 8.3 ± 0.6 Bev/c, one event of generation and decay of a Σ^- -hyperon was detected (see accompanying color photo and explanatory diagram).

The π -meson primary (track 1) interacts with a carbon nucleus to form four charged high-energy particles (tracks 2, 5, 7, 16), two K^0 -mesons (tracks 4, 5 and 14, 15), one low-energy particle (the short track 17), and the recoil nucleus.

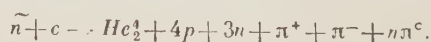
The decay of particle 2 at point A to particle 3 and a neutral particle in the direction AB is in excellent agreement with the kinematics of Σ -hyperon decay. Track 3 is the track of a π^+ -meson. The neutral particle at a distance of 7.7 mm from the point where the decay event occurred forms, at point B, a high-energy six-pronged star (tracks 8-13).

The energy contributed solely by the charged particles (1483 ± 60 Mev) is much higher than the kinetic energy of the neutral particle (940 ± 100 Mev). The neutral particle was thereby determined to be an antineutron. Under the assumption that a fraction of the energy was contributed by neutrons and π^0 -mesons, the energy in the star, after taking the binding energy

Diagram of the principal tracks showing on the photograph.

of nucleons within the nucleus into account, was determined to be higher than 2300 Mev. This energy is close to the annihilation energy of an antineutron.

The most probable reaction at point B would be:



This implies that the primary π^- -meson formed, at point 0, a hitherto unobserved particle, an antisigma minus hyperon, decaying at point A in accord with the decay scheme $\tilde{\Sigma}^- \rightarrow \pi^+ + \tilde{n}$. The presence of two neutral K-mesons allows us to write the reaction for the generation of the $\tilde{\Sigma}^-$ -hyperon as follows:

$$\pi^- + c \rightarrow \tilde{\Sigma}^- + K^0 + \bar{K}^0 + K^- + p + \pi^+ + \pi^- + \dots$$

The negative particle (track 6) is determined to be a K^- (by considerations of strangeness conservation).

For the lifetime of this $\tilde{\Sigma}^-$ -hyperon, the following value was arrived at:

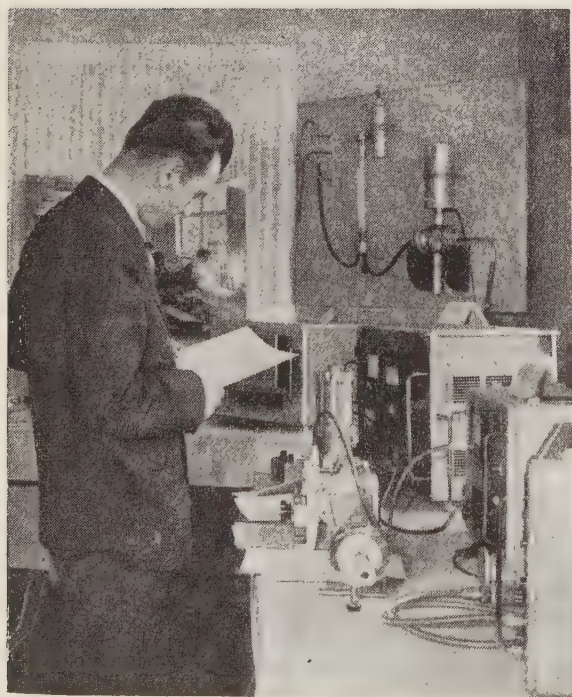
$$t_{\tilde{\Sigma}^-} = (1,18 \pm 0,07) \cdot 10^{-10} \text{ sec.}$$

This detection of the first charged antihyperon is a fact of enormous scientific significance. Our concepts of the microcosmos are further enriched thereby. One more stage has been reached in the knowledge of the nature of elementary particles and their transformations.

The discovery of the new particle is a great success for the entire staff of the High Energies Laboratory of the Institute.

HUNGARIAN EXHIBIT OF INSTRUMENTS FOR EXPERIMENTAL NUCLEAR PHYSICS RESEARCH

Translated from Atomnaya Énergiya, Vol. 8, No. 6, pp. 563-564,
June, 1960



At the Hungarian exposition on instruments for experimental nuclear physics. (Photo by M. Pyatkin).

In April 1960, the Hungarian exposition of instruments for experimental nuclear physics, organized by the Atomic Energy Commission of the Hungarian Peoples Republic and the Hungarian METRIMPEX foreign trade organization, was held at the Dubna Joint Institute for Nuclear Research. The exposition was dedicated to the 15th anniversary of the liberation of Hungary from the fascist aggressors.

In recent years, instrument design, and in particular the manufacture of instruments for nuclear physics applications, has developed in rapid stride in Hungary. It suffices to state that at the present time the volume of instrument production is 13 times that in 1950, and will have increased by 28 times in 1965.

Manufacture of mass-production instruments for experimental nuclear physics is being stepped up. To improve the quality and lower costs of items manufactured in enterprises engaged in mass production of electronic equipment, small research teams have been set up.

Instrument design is the concern of such large-scale scientific centers as the Central Scientific Research Institute for Physics attached to the Hungarian Academy of Sciences, and the research departments in the universities and the various institutes.

Instruments, diagrams, and wiring layouts were exhibited in the two halls occupied by the exposition (see accompanying photo). On demonstration here were a thickness gauge, automatic sample changer, timing controls, scalers, counting rate meters, a differential discriminator with resolving time of $5 \mu\text{sec}$, stabilized high-voltage power supplies, general-purpose scintillation detectors, dosimeters, γ -scintillation recording charts with image size 297 by 420 mm, affording physicians the possibility of studying a tissue by enriching the tissue with a radioactive isotope. Other counting and measuring instrumentation was also on display.

The electronic equipment is used in Hungary not only for scientific work in the field of the physics of the atomic nucleus, but also finds applications on a broad scale in medicine, agriculture, ferrous metallurgy, the food processing industry, and the plastics industry, and others.

The exhibition organized at Dubna showed the significant achievements of the Hungarian Peoples Republic in the peaceful uses of the gains of science and technology.

RECENT DATA ON C^{14} CONCENTRATION IN THE ATMOSPHERE

Yu. V. Sivintsev

Translated from *Atomnaya Énergiya*, Vol. 8, No. 6, pp. 573-575,
June, 1960

One of the scientific problems subjected to broadest discussion in recent years is the question of radiation hazards due to testing of nuclear weapons [1-4]. It is a familiar fact that an enormous quantity of neutrons is liberated in the explosion of any nuclear bomb, the neutrons going on to interact with nitrogen contained in the air, forming the radioactive isotope of carbon C^{14} :



Being a pure β -emitter, C^{14} decays with a half life of about 5,600 years, thanks to which the quantity of C^{14} in the atmosphere has been steadily increasing since 1953.

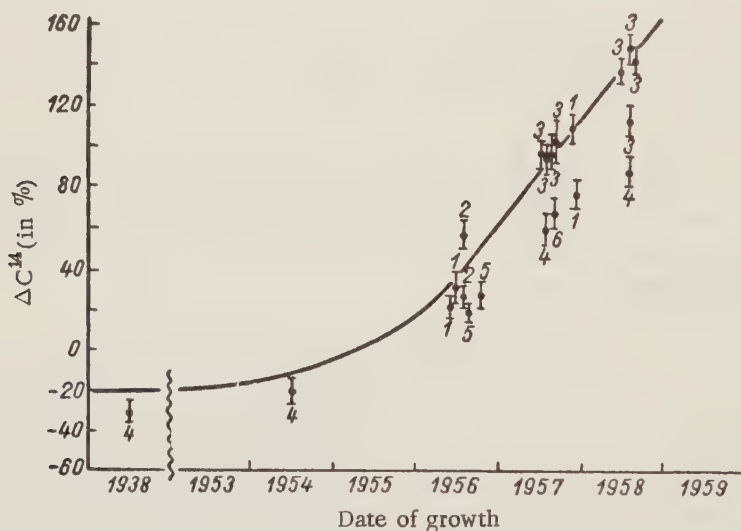
In contrast to atomic and uranium fission bombs, where the basic radiation hazard is associated with the concentration in one area of long-lived radioactive fragments, Sr^{90} in the first instance, taken into the human organism with the resultant formation of malignant neoplasms, the pure hydrogen (deuterium-tritium) fusion bomb is dangerous to both contemporary and future generations because of the formation of mutant genes in response to irradiation of the gonads by β -particles emitted by the isotope C^{14} . Assuming a linear dose dependence of the genetic effect, and also assuming that the spontaneous rate of mutation in humans is 10% due to natural irradiation, O. I. Leipunskii reached the conclusion [1] that the total number of genetic victims from the explosion of a 10-megaton pure hydrogen bomb comes to 49,000 persons, as against 41,000 persons for an explosion of a conventional nuclear bomb. In line with these data, it is interesting to quote U. S. statistics on the total equivalent of the nuclear test explosions carried out, which totals 174 megatons up to December 31, 1958, according to [5].

This reference [5] also gives the results of measurements of C^{14} concentration in atmospheric carbon dioxide gas over the past 4 years (1956-1959). The procedure used in the study consisted in direct mass-spectrometric determinations of the C^{14}/C^{12} ratio in samples of atmospheric carbon dioxide and plant substances which had assimilated carbon not long prior to the sampling measurements (ring growths on trees of several years age). For purposes of comparison, the results were normalized to the usual C^{13}/C^{14} ratio, which successfully eliminated the differences associated with a possible dilution of the isotopes in the process of laboratory treatment of the samples, or during photosynthesis (in the case of plant samples). The "background" content of C^{14} in wood was also taken into account in the normalization. The final result of measurement of a sample ΔC^{14} thus showed

the amount of radioactive carbon formed solely in the wake of nuclear weapons tests [6]. The results of the measurements for the Northern hemisphere [5], expressed in tenths of a percent, are diagrammed. These data indicate that the total annual increase in the long-lived radioactive C^{14} isotope present in atmospheric carbon dioxide during the last three years was about 5‰.

The data compiled increase the results of preceding investigations published in prior years. For example, one report [7] stated an annual increase of 4.3‰ for the Northern hemisphere, while another gave figures of 3.2‰ for Central Europe and 2‰ for South Africa [8].

The authors of [5] indicate some possible reasons for the low results in preceding reports. One is related to the dilution of atmospheric carbon dioxide by the enormous wastes of combustion products from fossil fuel in large-scale industrial regions. Because the C^{14} of natural origin present in those substances had undergone considerable decay, the air in large cities and their environs has a lower C^{14} content than in agricultural regions (this is termed the Suess effect [9]). To take this into account, the authors of [5] plotted a curve (see diagram) for samples collected over areas remote from industrial regions, above the Atlantic ocean (in the Northern hemisphere), and in the Mediterranean. It is interesting to note that the data reported in [8] for plant samples from industrial regions in West Germany showing a smaller annual increase in the amount of C^{14} display excellent agreement with samples taken in New York, Rome and Kearny (state of Nevada, USA) when plotted on the graph.



C^{14} concentration in carbon dioxide (Northern hemisphere) [5, 7-9]:
 1) Atlantic (atm. CO_2); 2) Mediterranean (atm. CO_2); 3) Great Plains (plants); 4) New York (plants); 5) Rome (plants); 6) Kearny, Nevada, USA (plants).

A second factor responsible for the harvest of low results in preceding years is failure to treat the effect of carbon dioxide carried by the soil. Since plants growing on high-yield acreage may obtain a considerable quantity of carbon dioxide from the soil, and since the latter probably formed through decay of organic substances biosynthesized prior to nuclear weapons testing, the difference between the C^{14} concentrations in soil and atmospheric carbon dioxide widens its span rapidly with the passage of time. Also of interest is the fact that, in the words of the authors of [5], the points used to plot the curve were obtained from sparsely sown areas and are therefore free from effects of soil carbon dioxide.

Comparing their results with the data presented in [10] for the Southern hemisphere, the authors of [5] stated that, although the overwhelming bulk of the weapons tests of recent years were carried out in the Northern hemisphere, the C^{14} concentration in the atmosphere of different hemispheres differs by at most 36%. This last result confirms the supposition advanced in [11] on the high rate of mixing of the atmosphere of the two hemispheres (mixing period of about two years). In conclusion, the authors of [5], after discussing the various possible mixing models for the stratosphere and troposphere of the two hemispheres, estimated the total quantity of bomb-

produced radioactive C^{14} accumulated up to March 1958 (see Table). One fact of extreme interest is that the most probable result which they arrived at coincides beautifully with the theoretical predictions put forth in [12].

As we see from the data presented in the table, the principal source of the uncertainty is the estimate of the amount of C^{14} accumulated in the stratosphere. Reference [5] is successfully supplemented by the investigations of stratosphere-borne C^{14} concentrations reported on in [13]. The authors, following a detailed description of the procedure used in sampling air at different altitudes (from 14 to 28.5 km) and measuring activities, present their empirical data on the quantity of A-bomb C^{14} present for four altitude belts and five sampling points, covering the period from 1958 to May 1959. The sampling points were spaced in a manner aimed at en-

Amount of Carbon-14 of A-bomb Origin (and H-Bomb Origin) Accumulated by March 1958, in 10^{27} C^{14} atoms.

	Tropo- sphere	Earth bio- sphere	Ocean	Strato- sphere	Total
Minimum.	3.6	0.2	0.6	0.7	5.1
Most probable value.	3.6	0.2	1.0	7.0	11.8
Maximum.	3.6	0.2	1.5	28.5	27.8

compassing a broad latitude belt (from 45°N to 25°S) and thus to obtain the material needed for resolving the question of the rate of mixing of the atmosphere between the two hemispheres. On the basis of these measurements, the authors asserted that the amount of C^{14} in the stratosphere due to bomb testing, over the period from 1955 to 1958, remained virtually constant (to an accuracy of $\pm 30\%$) and amounted to an average of $7.2 \cdot 10^{27}$ atoms of C^{14} with fluctuations from $5.6 \cdot 10^{27}$ to $8.6 \cdot 10^{27}$. However, Hagemann and associates [13] feel that since 85% of the nuclear explosions in recent years were conducted in the air, the fraction of the C^{14} formation and build-up refer-

able to each megaton exploded should be scaled upwards. On this basis, they arrived at the conclusion that the total quantity of C^{14} formed as of October 31, 1958 as a result of the detonation of nuclear weapons amounted to $25 \cdot 10^{27}$ atoms of C^{14} . If we assume this carbon to be evenly distributed over the entire atmosphere, then the total tropospheric concentration of C^{14} will be found to increase 1.75 times over the background estimate. It should be particularly stressed that the infiltration of atmospheric C^{14} into ocean waters, on which great hopes had been laid in earlier years, from the standpoint of a relatively rapid cleansing of the atmosphere, is a rather slow process, as seen in the light of the latest data. In particular, the authors of [5], basing their views on experimental determinations of the amount of C^{14} found in the surface waters of the Northern and Equatorial Atlantic, reported that only 10% of the C^{14} generated by bomb tests by March 1958 had found its way into ocean waters.

In conclusion, it should be mentioned that reference [14] gives the first three measurements of C^{14} concentration in tissues of the human organism (lungs, blood, respiratory acid) as a result of which they found that the C^{14} concentration in human tissues lags behind the C^{14} content of the atmosphere by as much as 1.1-1.8 years.

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*Original Russian pagination. See C. B. translation.

APPLICATIONS OF ALPHA RADIATION FROM RADIOACTIVE ISOTOPES FOR QUALITY CONTROL IN GRINDING OPERATIONS

V. V. Kondashevskii, A. N. Chertovskii,

V. S. Pogorelyi, and A. M. Gutkin

Translated from *Atomnaya Énergiya*, Vol. 8, No. 6, pp. 576-578, June, 1960

A transducer operating on the basis of the number of particles reaching the counter as a function of the transverse cross section seen by the rays of particles has been developed by the authors and has been studied under laboratory and production conditions. The application for which the transducer is intended is to provide new automatic controls for the grinding machine operation, with improved precision and reliability in performance.

The basic layout of the radiation transducer is shown in Fig. 1. The rod 2 of the transducer is held against the measuring rod 1 of the gage, which comes in contact with the part being monitored. Resting on the rod 2, and loaded by a spring 3, is an angle lever 4 connecting to a slide valve 5, which is inserted between the isotope 6 and a MST-17 type end-window particle counter 7. The radiation source is housed in a casing 8. Displacement of rod 1 results in rotation of the cranked lever and slide valve. The intensity of the α -radiation arriving at the end-window counter depends on the extent to which the slit opening in the valve orifice 9 is open.

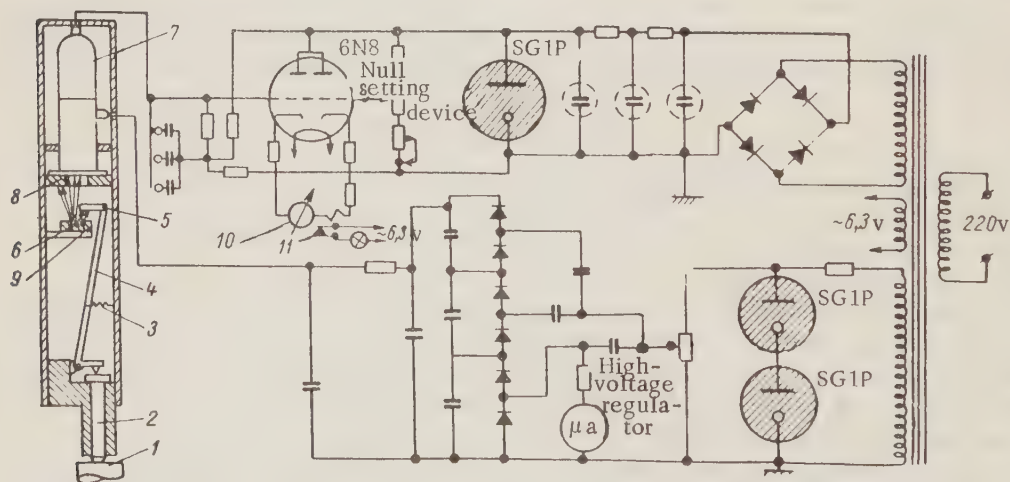


Fig. 1. Mechanical layout and electric circuitry of the radioisotope transducer.

The counter, whose function is to record the intensity of radiation from the isotope, is included in the electric circuit with a data-indication dial 10 and relay 11.

Any change in the size of the machined part being monitored can be sensed by reference to the dial; at the instant when the part is reduced to a predetermined size, the relay sends a command to automatically halt

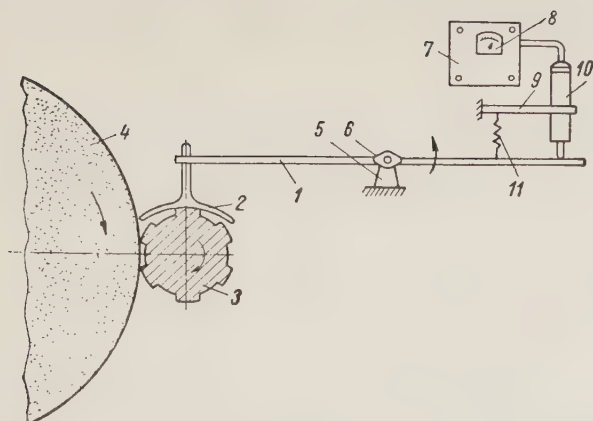


Fig. 2. Diagram of leverage design in radioisotope transducer for quality control of splined shafts: 1) connecting levers (of instrument); 2) arc-shaped measuring adapter; 3) part being machined; 4) grinding wheel; 5) link fulcrum; 6) axis of rotation; 7) faceplate for electrical assembly; 8) data indicator; 9) transducer support bracket; 10) radioisotope transducer; 11) spring.

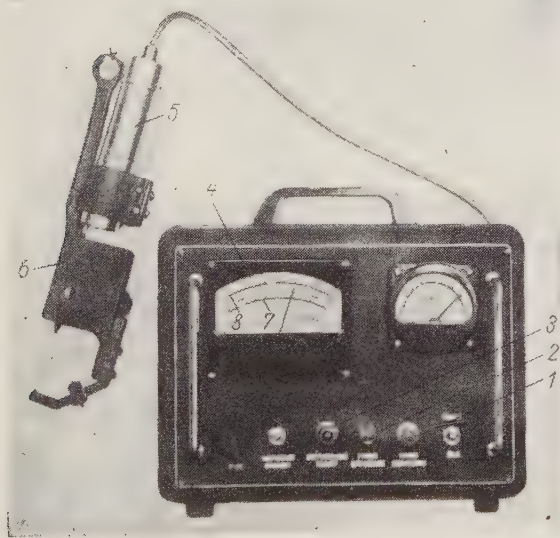


Fig. 3. Indicator-dial snap gauge with multicommand radioisotope transducer.

the machining process. Intensive research and development work bore fruit in optimization of the design parameters of the transducer components: slit cross section (in orifice) of 0.4 by 15 mm; transfer ratio of mechanical portion of transducer within the range 4 : 1 to 10 : 1.

The emitter selected was a thorium isotope for which the body of the transducer provided adequate shielding for the operator from emitted alphas. Only one paper* is known to the authors where an alpha-emitting radioisotope transducer has been mentioned.

The radiation transducer was mounted in a triple-contact snap gauge 6 (see Fig. 3) to monitor smooth cylindrical shafts during grinding, and was tested under laboratory conditions, followed up by later testing under production conditions on a grinder in the machine shop of the Omsk Sibzavod plant. Working under a tolerance of 23 microns, the spread in dimensioning of the machined shafts was found to be 13 microns, whereas use of a conventional gauge meets with a hard time in keeping within tolerances in shaft work.

In work with a single-point gauge (Fig. 2) coupled to the radioisotope transducer used to monitor grinding of splined shafts to an O. D. of 72 mm and to a tolerance of 17 microns, the actual scatter in the dimensions of the ground shafts was found to be 17 microns. Without the aid of an instrument, a skilled operator would encounter difficulty in machining the parts to within tolerance specifications.

The electrical circuitry for the radioisotope transducer, described above, requires only one command in feeding work to the grinder, the command to retract the wheel from working position. In some cases, this one command turns out to be inadequate. The radioisotope transducer 5 shown in Fig. 3 is built to give three commands to a machine tool:

1. Command to switch from rough feed to finish feed, when the machined part has been dimensioned down to 30-60 microns within specifications. This command switches on an annunciator light bulb 1.

2. Command to stop finish feed when the part is machined to 10-15 microns of specification. Light bulb 2 switches on at this point and the final finish machining is then initiated without feeding the wheel into the work ("coasting").

3. A command to instantaneously retract the grinding wheel when the final dimensioning is completed. Light bulb 3 flashes on at this point.

*M. B. Neiman. Use of radioactive isotopes in machine design [in Russian]. Symposium "Automation of manufacturing processes in machine building (control applications)". Moscow, Academy of Sciences Press, 1955.

The indicator snap gauge 4 has two reaching scales: one coarse scale 7 and a precision scale 8. The scale divisions of the precision scale run from 0.5 to 2 microns (set during adjustment), and the divisions of the coarse scale run from 2.5 to 10 microns. The dial scales are switched on and off automatically. The presence of two scales facilitates observation and monitoring of changes in part size.

Comparative tests of radioisotopes, inductive, pneumatic, and electric-contact transducers have shown that the precision of the radioactive transducers matches the levels of the best variable inductors. The cost of the radioisotope transducer and the complexity of its electrical circuitry do not exceed the cost and complexity of variable inductors.

BRIEF COMMUNICATIONS

Translated from *Atomnaya Énergiya*, Vol. 8, No. 6, p. 578
June, 1960

USSR. In Minsk, construction work was completed on the main building of the 2000 kw (th) research reactor of the Academy of Sciences of the Byelorussian SSR. Assembly of the reactor core and equipment for beams of radiation is in progress. Electrical engineering equipment, control and measuring instrumentation and automatic controls are being put in place.

The reactor is designed for biological and miscellaneous research projects, production of radioactive isotopes, and studies of the behavior of various materials under exposure to neutron and gamma-ray bombardment.

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NEW LITERATURE

Translated from *Atomnaya Énergiya*, Vol. 8, No. 6, pp. 581-583,
June, 1960

BOOKS AND SYMPOSIA

N. F. Nelipa, *Coupling Between Photoproduction of π -mesons and Scattering*. Moscow, Atomizdat, 1959
88 pages, 4 rubles.

This text gives a systematic review of papers on coupling between photoproduction of π -mesons on nucleons and scattering, and the application of the results obtained to an analysis of experimental data. The book is supplemented with Tables of Clebsch-Gordon, Racah, and Z coefficients.

The book is written for scientific research workers engaged in the study of nuclear reactions at high energies, and for students taking nuclear physics programs.

Controlled Thermonuclear Reactions. A symposium of translated materials. Moscow, Atomizdat, 1960.
319 pages. (Main control board on atomic energy. Directors of science and engineering information and exhibits. No. 26). 14 rubles, 80 kopeks.

This symposium contains 20 articles reflecting the level of work on controlled fusion in Britain and West Germany at the start of 1957. The burden of the symposium centers on two series of articles dealing with the work of a British group at Harwell and a German group at Göttingen. Work by American physicists was not readily available at that time, and is represented in only very limited degree. The symposium concentrates mainly on theoretical research, and experimental work is represented by several brief communications.

The symposium will prove useful for persons interested in plasma physics and the physics of controlled thermonuclear reactions.

J. L. Synge, *Relativistic Gas*. Translated from the English, edited by D. A. Frank-Kamenetskii. Moscow, Atomizdat, 1960, 140 pages, 4 rubles, 20 kopeks.

Operation "Argus". Translated from the English. Moscow, Atomizdat, 1960, 160 pages, 6 rubles.

This book constitutes a symposium of papers presented at the special symposium on operation "Argus", held in April 1959.

Light is shed on the results of observations of the behavior of electrons trapped by geomagnetic fields. The observations were conducted with the aid of artificial earth satellites during a series of nuclear explosions carried out at heights of 480 km by the USA, in 1958.

The book will be of interest to physicists, astrophysicists, and meteorologists in the first instance. However, it is also within reach of a broader readership.

D. Ya. Surazhskii, *Techniques in Prospecting and Exploration of Uranium Deposits*. Moscow, Atomizdat, 1960, 240 pages, 8 rubles, 70 kopeks.

This book is a handbook of techniques on one of the most important branches of prospecting and explorative geology, compiled on the basis of the experience accumulated in prospecting and exploring uranium deposits in the USSR and abroad.

The book also provides a general picture of the types of uranium deposits which are of interest, and the main criteria governing prospecting activities are treated. Methods for prospecting uranium beds by radiation, gas, and salt haloes are described, as well as methods of preliminary prospecting, detailed exploration, and sampling of ore bodies.

The book is intended for geologist-engineers, technicians, and students in institutes teaching mining, geology and exploration geology.

Kh. B. Mezhiborskaya, V. L. Shashkin, and I. P. Shumilin, *Analysis of Radioactive Ores by the beta-gamma-Technique*. Moscow, Atomizdat, 1960, 64 pages, 1 ruble, 90 kopeks.

This text is devoted to radiometric analysis of samples of uraniferous and uranium-thorium ores with disturbed radioactive equilibrium.

Topics discussed include the theory of the β - γ -method, the equipment used, special problems in measurement of β - γ -emission from samples, methods for determining the coefficient in computations and formulas. Recommendations are given for estimating the accuracy of analyses made by the β - γ -technique.

A short description is given of methods of radiometric analysis of samples in complex radioactive ores.

The book is written for physicists and geophysicists working in the field of the analysis of radioactive ores. It may also be found useful by students in the corresponding specialties, as a manual for use in a course on radiometry.

Extraction and Purification of Exotic Metals. Translated from the English, edited by O. P. Kolchin, Moscow, Atomizdat, 1960, 512 pages, 24 rubles, 35 kopeks.

This book consists of a collection of papers presented at the 1956 symposium of the London Institute of Mining and Metallurgy.

The 22 papers give the results of laboratory research, and in some cases of industrial research, on the technology of uranium, thorium, beryllium, zirconium, hafnium, niobium, vanadium, titanium, selenium, and several other rare metals.

The book will be of interest to metallurgical engineers, chemical engineers, ore processors, and scientific research workers engaged in the field of the production and application of radioactive and exotic metals.

S. V. Elinson and K. I. Petrov, *Analytical Chemistry of Zirconium*. Moscow, Atomizdat, 1960, 212 pages, 7 rubles, 80 kopeks.

The chemical and physical-chemical properties of zirconium and zirconium compounds are discussed. The most important analytical reactions and methods for detecting zirconium in other materials are described. Methods for isolating zirconium from the other elements are discussed. A detailed exposition is given of the volumetric, gravimetric, calorimetric, and spectral techniques for zirconium assay in alloys, salts, and other materials. Techniques for determining gas-forming elements and carbon present in zirconium are dealt with extensively, and the chemical and spectral techniques for determining other trace impurities in zirconium and components present in zirconium alloys are also discussed.

This text may be used as a practical handbook for workers in plant laboratories and research institutes, and also as a textbook manual for students in chemical and metallurgical institutes.

M. I. Shal'nov, *Neutron Tissue Dose*. Edited by B. M. Isaev, Atomizdat, 1960, 218 pages, 8 rubles, 20 kopeks.

This book represents an attempt to generalize the data culled from the literature, as well as the materials obtained by the book's author in his own experimental work with neutrons, on all of the principal questions involved in tissue dosimetry. The first two chapters are devoted to the basic properties and sources of neutrons, and to the interaction of neutrons with matter. Subsequent chapters discuss the theoretical and experimental results on depth distribution of absorbed neutron dose in tissue-simulating media; the principal problems associated with the study of the relative biological effectiveness of nuclear radiations, the problem of the maximum tolerable dose, etc. are discussed. The last chapter contains a concise review of the instruments and techniques of neutron dosimetry.

This book is of interest to biologists and medical workers engaged in the field of radiation biology and radiation medicine.

Chemical Protection of the Organism from Ionizing Radiations. Edited by V. S. Balabukh. Moscow, Atomizdat, 1960, 152 pages, 4 rubles, 30 kopeks.

The first part of this symposium contains a brief review of the problems encountered in chemical protection from ionizing radiations, experimental data on the synthesis of, and biological testing of the protective properties of a number of chemical compounds.

The second portion deals with the results of experimental research on isolation of radioactive isotopes from the organism. The characteristics of the state of several radioactive isotopes in the blood and bone tissue are discussed. The physical-chemical and biological evaluation of the effectiveness of complexing agents used to remove isotopes from the organism are discussed.

The symposium is intended for chemists engaged in the field of searching for means of chemical protection, and for complexing agents capable of serving that function, as well as for those biologists and other specialists concerned with radiation biology problems.

G. O. Davidson, Biological Consequences of Total-Body Irradiation in Humans. Translation from the English. Edited by M. F. Popova. Moscow, Atomizdat, 1960, 108 pages, 4 rubles, 70 kopeks.

The chief task of this book is an attempt to draw conclusions of practical interest on shielding against the effects of radioactive precipitates, based on biological and medical data on the effects of ionizing radiations on animal and human organisms.

Experimental material on the relationship of lethal doses in instantaneous, intermittent, and chronic irradiation is discussed. The theory of injury and recovery following radiation injury is developed, and attention is given briefly to the immediate sequelae of irradiation. The experimental material given points up the pathways of possible future biological research having a direct bearing on problems of shielding against radioactive ionizations.

The book is written for a broad readership of biologists and physicians interested in problems of radiation shielding.

G. Shreiber, Biophysical radiology. Translated from the German. Moscow, Atomizdat, 1960, 368 pages, 18 rubles, 50 kopeks.

This book is an outline of a course of lectures presented by the author at the Humboldt University in Berlin, as an introduction to radiology. The book discusses the fundamental concepts of the physics of ionizing radiations; presents data on the most important physical, physico-chemical, and biological effects occurring in response to the interaction of radiation with matter; and sheds light on questions concerning dosimetry.

The book is written primarily for biologists and medical workers working in the fields of radiology radiation selection, etc.

L. S. Kozyreva-Adelsandrova, and N. I. Temnikova, The Radioactive Isotope Iodine-131. Moscow, Atomizdat, 1960, 24 pages, 50 kopeks.

This brochure describes the properties of radioactive iodine and discusses the various fields of application for the isotope. Concrete examples of the use of I^{131} in medical practice for diagnostic procedures and therapeutic applications, in chemistry for the study of chemical processes, etc. are given. Recommendations on rules for handling I^{131} and health physics questions are treated. The introductory part of the brochure cites briefly the characteristics of the isotopes and techniques used in its production.

The brochure is written for a broad audience.

N. P. Galkin, A. A. Maiorov, and U. D. Veryatin, Technology of the Processing of Uranium Concentrates. Moscow, Atomizdat, 1960, 162 pages, 6 rubles, 50 kopeks.

The book gives a short outline of the development of the uranium industry. Fundamental information on hydrometallurgical processes for isolating uranium from the raw ore material, on reserves of uranium ores,

scales of production, and fields of application for uranium then follow. The bulk of the attention is given over the technology of processing of uranium concentrates to pure salts and uranium metal.

Methods for producing the most important uranium compounds are described, and their physico-chemical properties are cited. Production flow charts used in various countries for uranium metal production processes are compared. A special chapter deals with problems of safety and sanitation in the process of purification of uranium chemical concentrates.

The book is written for workers in the uranium industry, scientific research organizations, and may be used for the training of corresponding specialists in advanced institutions.

S. V. Rumyantsev, *Radioactive Isotope Applications in Nondestructive Testing*. Moscow, Atomizdat, 1960, 296 pages, 11 rubles, 25 kopeks.

The physical and engineering fundamentals concerned in applications of radioactive isotopes in nondestructive testing for flaws in materials are outlined in this book. Characteristics of the isotopes and their fields of application are cited. The procedure used in radiographing manufactured parts and handling the associated equipment is discussed. Problems related to the effect of metallurgical flaws on the strength of weld joints are investigated. A presentation is made of the ionization technique, of xeroradiography, etc. The problems of safe handling of isotopes, shielding from ionizing radiations, are elucidated.

The book is written for engineers and technicians engaged in flaw detection and nondestructive testing.

L. K. Tatochenko, *Radioactive Isotopes in Instrument Design*. Moscow, Atomizdat, 1960, 368 pages, 13 rubles, 20 kopeks.

The theoretical and practical problems related to the use of radioactive isotopes in industry are discussed in this book. The procedure followed in the engineering design of instruments based on the use of radiations is discussed. Light is shed on problems concerning organization of work with and handling of radioactive isotopes, concerning the dosimetry of ionizing radiations, and radiation safety practices.

The book is written for engineers and scientists working in various fields of industry.

English-Spanish-Russian-French Dictionary of Scientific and Engineering Terms on Nuclear Energy. Moscow, Atomizdat, 1959, 215 pages, 16 rubles.

The dictionary is reproduced from the fourth edition of the English-Spanish-Russian-French dictionary of scientific and technical terms on atomic energy published by the Terminology Section of the UN in 1958. The dictionary contains about 6,000 terms in four languages. The material is arranged in alphabetical order according to English entries. To facilitate searching for Spanish, Russian, and French terms, the dictionary has special index sections appended.

The dictionary was compiled for the benefit of scientists, engineers, translators, and students in the area of interest.

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278	left	14-18	The increase . . .	The increase in the magnitude of the longitudinal velocity component $v_{\parallel} \approx v_0$ to values $v_{\parallel} \approx v_{\perp} > v_0$ means that it is necessary to take account of the additional reduction of $\tau_c(E)$ due to the reduction in the time spent by the ion ($\tau_{\parallel}$) in the region of the heating section: $\tau_{\parallel} \approx 1 \text{ kv}_{\parallel}$.

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